





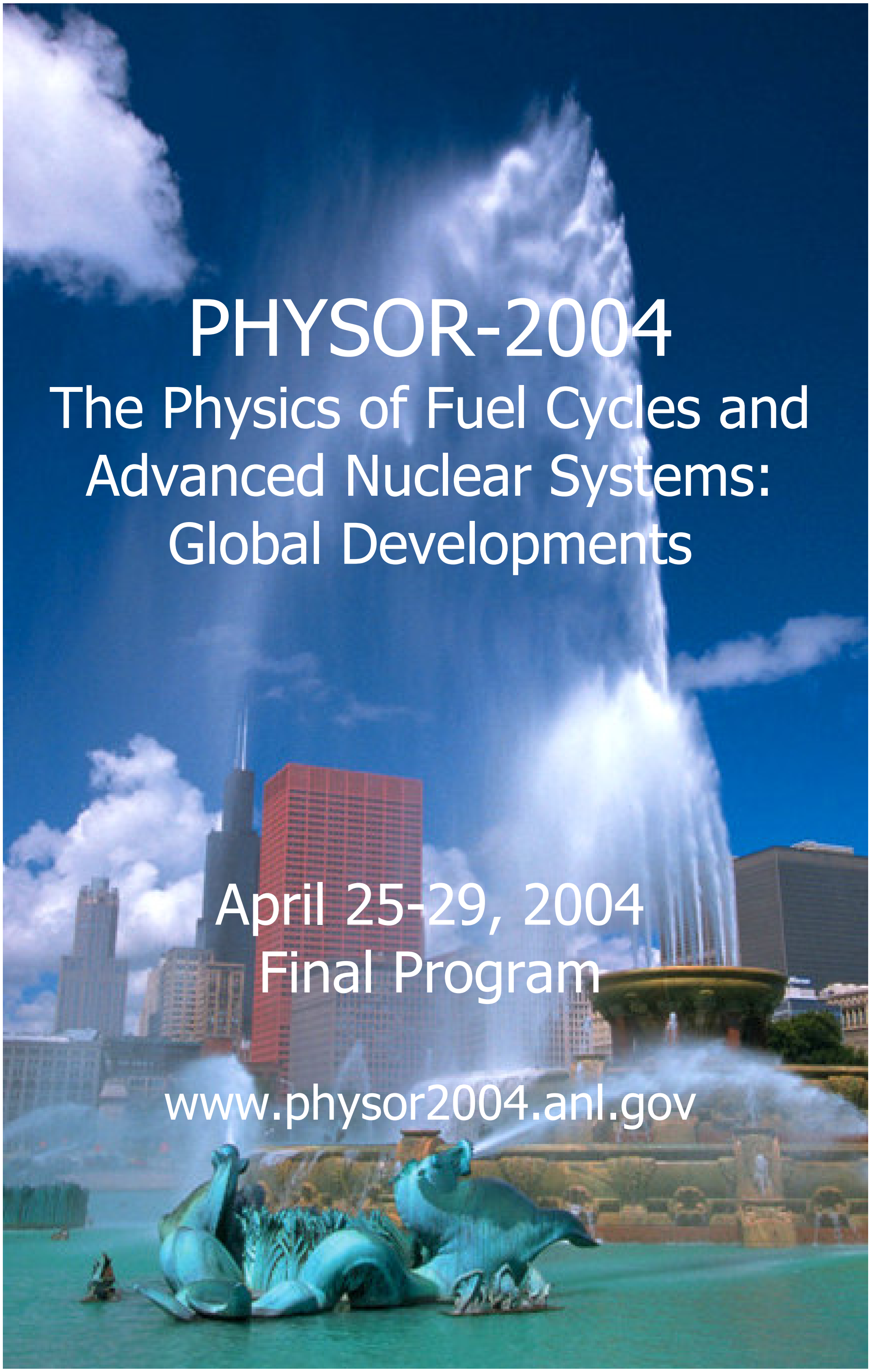

PHYSOR-2004

The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments

April 25-29, 2004

Final Program

www.physor2004.anl.gov

PHYSOR 2004 is a topical meeting sponsored by the Reactor Physics Division of the American Nuclear Society, and co-sponsored by the Mathematics & Computation Division. We also wish to express our sincere thanks to the following organizations for their support.

Non-financial support has been provided by:

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PHYSOR-2004 Topical Meeting

"The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments"

April 25-29, 2004
Chicago, Illinois, USA
Hyatt Regency Chicago

Final Program

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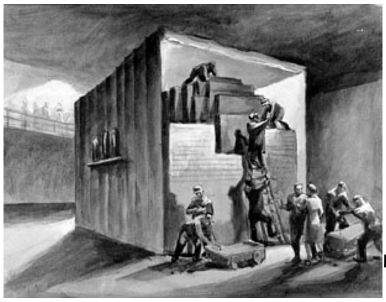
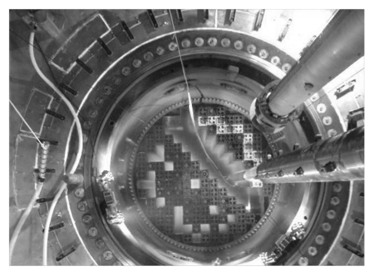


This meeting on the Physics of Reactors is one of a series of meetings sponsored by the Reactor Physics Division (RPD) of the American Nuclear Society (ANS). The purpose of the meeting is to provide a forum for reviewing recent developments in reactor physics and related computational methods and applications for nuclear power and associated technologies, identifying research and development needs, and setting the stage for future advances.

We welcome your participation at the PHYSOR-2004 Topical Meeting. Over 60 years ago, Chicago served as the birthplace of nuclear reactor technology. Today, the city provides a fitting location for the presentation of global developments in reactor and fuel cycle physics for mature technologies, and for a new generation of nuclear systems.



Best Regards,

Massimo Salvatores   Yoon Chang
General Chairs, PHYSOR-2004

Over sixty years ago, on December 2, 1942, the world's first self-sustaining nuclear chain reaction took place on a squash court beneath Stagg Field on the University of Chicago campus. Right after Enrico Fermi ordered the reaction stopped, the Hungarian born theoretical physicist Eugene Wigner presented him with a bottle of Chianti wine. Fermi uncorked the wine bottle and sent out for paper cups. He poured a little wine in each cup, and silently, solemnly, without toasts, the scientists raised the cups to their lips -- the Canadian Zinn, Compton, Anderson, Hilberry, and a score of others. They drank to success -- and to hope they were the first to succeed.

As the group filed from the West Stands, one of the guards asked Zinn: "What's going on, Doctor, something happen in there?" The guard did not hear the message which Arthur Compton was giving James B. Conant at Harvard, by long distance telephone. Their code was not prearranged.

"The Italian navigator has landed in the New World," said Compton.

"How were the natives?" asked Conant.

"Very friendly," was the reply.

U.S. Atomic Energy Commission, Washington, D.C., November 1949

Welcome from the Technical Program Chair

Dear Colleague,

The Reactor Physics Division of the American Nuclear Society (ANS) appreciates your participation in the PHYSOR-2004 Topical Meeting, held at the Hyatt Regency Hotel, Chicago, Illinois. This four-day meeting is hosted by the Chicago Section of the ANS and co-sponsored by many international organizations.

The theme of the meeting is *The Physics of Fuel Cycles and Advanced Nuclear Systems: Global Developments*. The technical sessions cover 21 separate topics, and 5 plenary sessions with internationally known speakers are also on the schedule. On the following pages, we are pleased to provide the agenda for the technical sessions and other special events. The scheduled date and time for each presentation are also provided.

In addition to the technical program, we want to remind you about three special events that are taking place on the days before and after the meeting.

- **MCNP Criticality Workshop**, Sunday, April 25, 1:00 to 5:00 pm. This free workshop will be held at the Hyatt Regency. The session will include an overview of MCNP criticality calculations and specific discussions of new features in MCNP5, including mesh tallies, the ENDF66 and SAB2002 nuclear data libraries, and generation of temperature-specific nuclear data libraries. In addition, there will be discussion of additional capabilities being developed for the next release of MCNP. Presenters will be Forrest Brown and Russ Mosteller from the MCNP team at Los Alamos National Laboratory. Meeting registration is required to participate in this workshop.
- **Welcome Reception at the Art Institute of Chicago,** Sunday, April 25, 6:00 to 9:00 pm. One ticket to this special reception is included with your conference registration. The reception will be held on Sunday evening, from 6 to 9 PM, and will include access to the Grand Staircase, the Gunsaulus Hall, and viewing of the Chagall Windows and the Impressionists and Post-Impressionists exhibits. The Art Institute is just a short walk from the Hyatt.



- **Generation-IV Reactor Physics Workshop,** Friday, April 30, 8:30 am to 5:00 pm. The intent of this workshop is to have international experts present/discuss ongoing R&D activities in the design and evaluation of the advanced nuclear systems. Core physics design and nuclear data needs and issues will be discussed along with proposals for international collaborations for meeting these needs. This all-day workshop will have presentations and discussion sessions. It will be held at the Hyatt Regency and meeting registration is required.

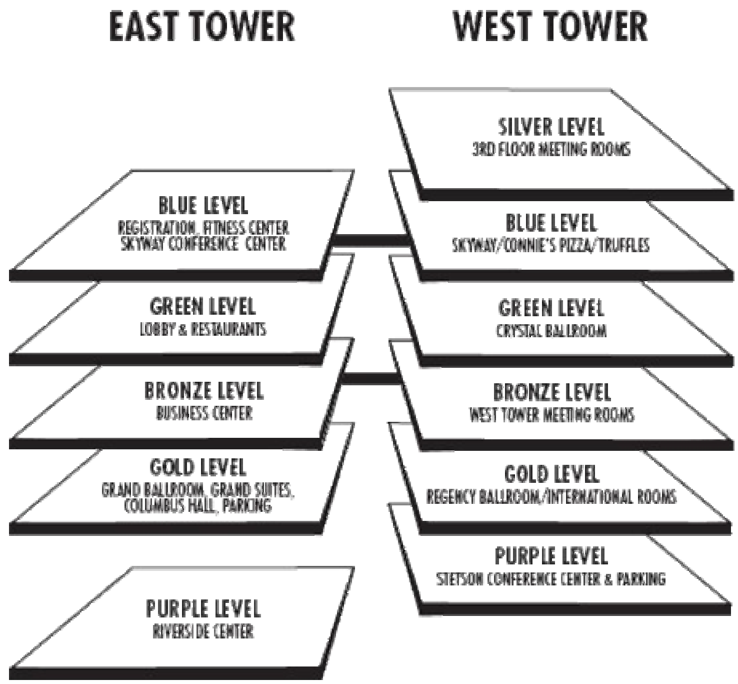
We encourage you to participate in these events as your schedule allows. We are looking forward to a full and informative meeting and hope you enjoy your stay in Chicago!

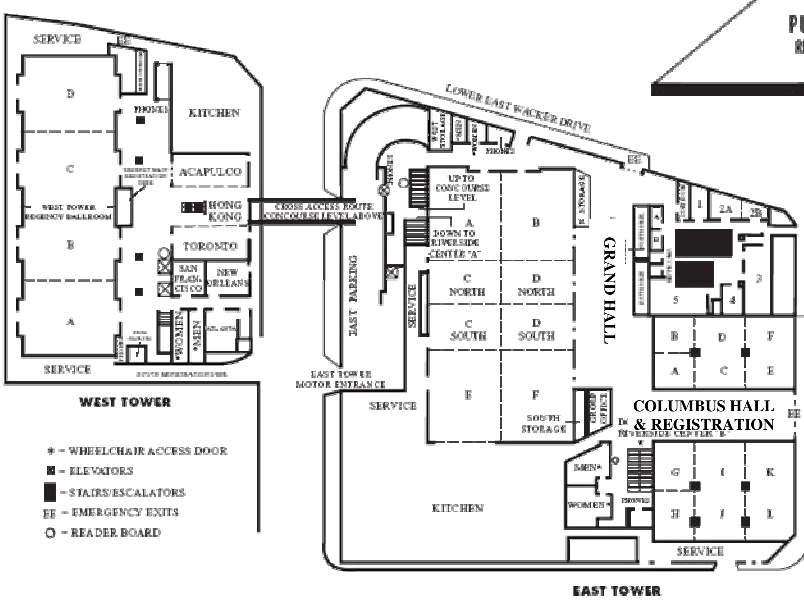
Best Regards,

P Finck

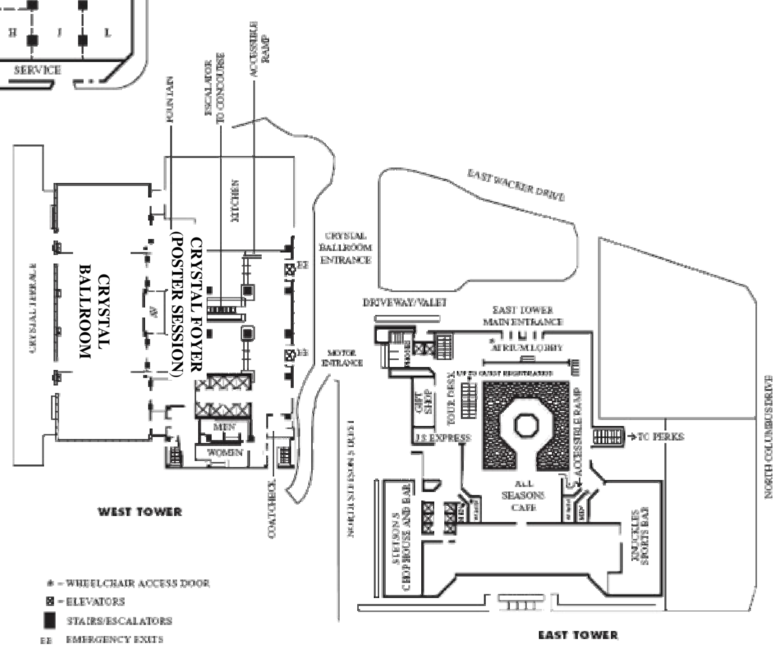
Phillip Finck, Technical Program Committee Chair, PHYSOR-2004

Registration, and all plenary, special, and technical sessions for PHYSOR-2004, will be located on the Gold level of the Hyatt Regency Chicago. The Tuesday evening Banquet Reception (& Poster Session) and Banquet will be in the Crystal Ballroom located on the Green level.





Gold Level –
Registration Area,
Grand & Columbus Halls



Green Level –
Crystal Ballroom & Foyer

PHYSOR-2004 Meeting Schedule

Day	Session	Time	Room	Session
Sunday, April 25		1:00 PM - 5:00 PM	Columbus EF	MCNP5 Criticality Workshop
		6:00 PM - 9:00 PM	Art Institute of Chicago	*Welcome Reception at Art Institute of Chicago*

Day	Session	Time	Room	Session
Monday, April 26		7:00 AM - 8:00 AM	Columbus IJKL	**Speakers' Breakfast**
		8:30 AM - 12:00 PM	Grand AB	**Plenary Session** - *International Perspectives in Reactor Physics*
		12:00 PM - 1:30 PM	Columbus IJKL	**Luncheon** - *Historical Perspectives on Reactor Physics Experiments*
	1A	1:30 PM - 4:50 PM	Columbus EF	International Collaboration in Reactor Physics
	1B	1:30 PM - 4:50 PM	Columbus G	Monte-Carlo Methods and Developments
	1C	1:30 PM - 4:50 PM	Columbus H	Reactor Physics Benchmarks and Experiments
	1D	1:30 PM - 4:50 PM	Grand B	Reactor Analysis Methods
		3:10 PM - 3:30 PM		**Break - All sessions**
		5:00 PM - 7:00 PM	Grand AB	**Special Session** - *Reactor Physics for Future Nuclear Systems*

Day	Session	Time	Room	Session
Tuesday, April 27		7:00 AM - 8:00 AM	Columbus IJKL	**Speakers' Breakfast**
		8:30 AM - 9:45 AM	Grand AB	**Plenary Session** - *Current Needs in Reactor Physics*
		9:45 AM - 10:00 AM		**Break**
	2A	10:00 AM - 12:00 PM	Columbus EF	Deep-Burn Physics and Methodology
	2B	10:00 AM - 12:00 PM	Columbus G	Accelerator Applications and Spallation Physics
	2C	10:00 AM - 12:00 PM	Columbus H	Reactor Physics and Materials Issues
	2D	10:00 AM - 12:00 PM	Grand B	Advances in LWR Analyses
		12:00 PM - 1:00 PM		**Lunch Break**
		1:00 PM - 2:15 PM	Grand AB	**Plenary Session** - *Experimental Activities in Reactor Physics*
		2:15 PM 2:30 PM		**Break**
	3A	2:30 PM - 4:50 PM	Columbus EF	Reactor and Neutron Physics
	3B	2:30 PM - 4:50 PM	Columbus G	Research Reactors
	3C	2:30 PM - 4:50 PM	Columbus H	Fuel Cycle Physics
	3D	2:30 PM - 4:50 PM	Grand B	Criticality Benchmarks
	4A	5:00 PM - 7:00 PM	Crystal Ballroom Foyer	Poster Session and Reception
		7:00 PM - 10:00 PM	Crystal Ballroom	**Banquet**

Day	Session	Time			Room	Session
Wednesday, April 28		7:00 AM	-	8:00 AM	Columbus IJKL	**Speakers' Breakfast**
	5A	8:30 AM	-	11:50 AM	Columbus EF	Fuel/Core Design and Analysis
	5B	8:30 AM	-	11:50 AM	Columbus G	Gas-Cooled Reactors
	5C	8:30 AM	-	11:50 AM	Columbus H	Critical and Subcritical Experiments
	5D	8:30 AM	-	11:50 AM	Grand D-South	Reactor Analysis Methods
		10:10 AM	-	10:30 AM		**Break – All sessions**
		12:00 PM	-	1:30 PM	Columbus IJKL	**Luncheon** - *Emerging Areas in Reactor Physics*
	6A	1:30 PM	-	4:50 PM	Columbus EF	Advanced Reactor Designs
	6B	1:30 PM	-	4:50 PM	Columbus G	Nuclear Safety
	6C	1:30 PM	-	4:50 PM	Columbus H	Physics Code Validation
	6D	1:30 PM	-	4:50 PM	Grand D-North	Reactor Analysis Methods
		3:10 PM	-	3:30 PM		**Break - All sessions**

Thursday, April 29		7:00 AM	-	8:00 AM	Columbus IJKL	**Speakers' Breakfast**
		8:30 AM	-	9:45 AM	Columbus IJKL	**Plenary Session** - *Efforts in Code Development and Reactor Modeling*
		9:45 AM		10:00 AM		**Break**
	7A	10:00 AM	-	12:00 PM	Columbus EF	Non-Conventional Reactors
	7B	10:00 AM	-	12:00 PM	Columbus CD	Nuclear Data
	7C	10:00 AM	-	12:00 PM	Columbus KL	Nuclear Safety
	7D	10:00 AM	-	12:00 PM	Columbus IJ	Physics and Modeling of Research Reactors in INIE's Big-10 Consortium
		12:00 PM	-	1:00 PM		**Lunch Break**
	8A	1:00 PM	-	3:20 PM	Columbus EF	Fuel Cycle Physics
	8B	1:00 PM	-	3:40 PM	Columbus CD	Nuclear Data
	8C	1:00 PM	-	3:20 PM	Columbus KL	Physics Code Validation
	8D	1:00 PM	-	3:40 PM	Columbus IJ	Reactor Analysis Methods

Friday, April 30		8:30 AM	-	5:00 PM	Columbus GH	Generation IV Reactor Physics Workshop

Registration

The PHSYOR-2004 Registration Desk is located at the Hyatt Regency on the Gold Level/East Tower, in the Columbus Hall Foyer. Meeting registration is required for all attendees and presenters. Badges are issued at Registration and are required for admission to all technical sessions, plenary sessions, special sessions, workshops, and special events. Extra tickets for the Sunday Evening Reception, Tuesday Banquet, and Monday and Wednesday Luncheons may be purchased at the Registration Desk. Extra copies of the Proceedings (distributed on CD-ROM) are also available for purchase.

Registration hours are as follows:

Sunday, April 25th
11:00 a.m. to 6:00 p.m.

Monday, April 26th
7:30 a.m. to 5:00 p.m.

Tuesday, April 27th
7:30 a.m. to 5:00 p.m.

Wednesday, April 28th
8:00 a.m. to 5:00 p.m.

Thursday, April 29th
8:00 a.m. to 12:00 p.m.

If you have any questions or special needs, please ask for assistance at the PHYSOR-2004 Registration Desk or in the meeting office located in the Skyway Conference Center, Suite 284 (Blue Level/East Tower).

Message Board

A message board will be located in the Registration area. Messages may be left for meeting attendees by calling 630-631-3785. E-mail messages less than 100 characters (including the subject line) can be sent to 6306313785@vmobl.com. Longer e-mail messages may be sent to physor.registration@anl.gov, but please note that this address will be checked only twice each day.

Welcome Reception

A special Welcome Reception will be held at the Art Institute of Chicago on Sunday evening, from 6:00 to 9:00 p.m. The Art Institute is located at 111 S. Michigan Avenue, approximately ½ mile south of the Hyatt Regency. Walking to the Art Institute is a pleasant 10 to 15 minute walk from the Hyatt. A downtown map is provided on page 8 of the Program.

Information for Speakers

All speakers and session chairs must sign in at the "speakers' desk" located near the Registration area.

Speakers are allotted 20 minutes for their presentation, including questions. Overhead and LCD projection equipment will be available, and laptop computers will be available in each meeting room. We suggest that you have your presentation on a CD-ROM or portable USB-drive in Adobe Acrobat (.pdf) or PowerPoint (.ppt) format and test the file on one of the computers ahead of your session time. In the event of an incompatibility of file formats, presenters should be prepared with an alternative (e.g. their own laptop computer or overhead transparencies). A speaker preparation room (for testing your presentation) will be located in the Skyway Conference Center, Suite 283 (Blue Level/East Tower).

A Speakers' Breakfast will be held each morning from 7:00 to 8:00 a.m. All speakers and session chairs for that day are invited to attend. The breakfast will be held in Columbus IJKL.

Workshops

The MCNP Criticality and Generation-IV Reactor Physics workshops will be held at the Hyatt Regency on Sunday afternnon and Friday, respectively. Both of these free workshops will be held in the Columbus Hall. Meeting registration and advance registration for each workshop are required. See pages 13 and 14 for more details.

Chicago Attractions

Many restaurants and other entertainment attractions are within walking distance of the hotel, or you can use public transportation or taxi service. The East and West Tower Doormen at the Hyatt Regency will be happy to secure a cab for you. Valet, overnight parking at the Hyatt with in/out privileges is $35; daytime parking is $16 (out by 7 p.m.). Self-parking lots are also located near the hotel. Chicago attractions include

- The Field Museum of Natural History, Shedd Aquarium, Adler Planetarium
- Navy Pier, Michigan Avenue, Water Tower Place
- Broadway in Chicago, The Chicago Theater, Second City Comedy Club

City of Chicago

The first resident of Chicago was Jean Baptiste Point du Sable, a fur trader from Santo Domingo of French-African descent. DuSable built the first settlement in 1779 at the mouth of the Chicago River.

In 1833, the Town of Chicago was incorporated, drawing its name from an Indian word meaning "strong" or "great".

On Oct. 8, 1871, a fire began on the West Side. Two days later, the Great Chicago Fire had claimed 300 lives, left 90,000 Chicagoans without homes and destroyed $200 million worth of property. The disaster was turned into an opportunity to plan and rebuild the entire city.

In 1893, Chicago hosted the World's Columbia Exposition that attracted nearly 26 million visitors during its six-month run. In order to provide transportation to the fair, the Chicago Transit Authority introducted the first elevated trains to Chicago. Today, the system's 'L' train encircles the city's central business district, referred to as "the Loop."

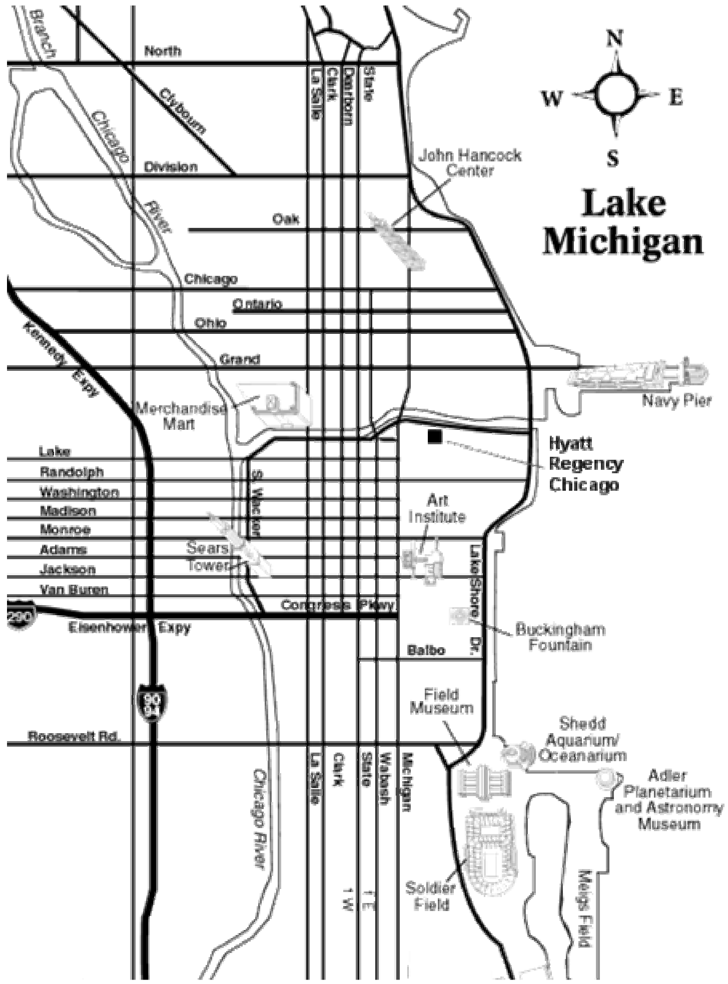
Chicago's cultural interests can be traced to this era, when its orchestra, library and major museums were established. The Columbian Exposition's Palace of Fine Arts is now home to the Museum of Science and Industry.

In 1909, the newly-formed Chicago Plan Commission published Daniel Burnham's comprehensive plan. The city's unobstructed lakefront, its city-wide system of parks and its green belt of forest preserves were all part of this unique plan, the first ever presented to an American city.

Chicago's multicultural heritage is reflected in its neighborhoods, which now attract thousands of visitors each year. Chicago is home to nearly three million people from all over the world including African Americans, Asians, Europeans, Hispanics, Native Americans and more.

As each new group has come to Chicago, their unique community spirit, typified by Chicago's motto "I will," has enabled them to build a new community, new life and new future. This spirit is responsible for a city that has never stopped dreaming, building, rebuilding, growing and making major contributions to the world.

The maps provided below show the Hyatt Regency, the Art Institute of Chicago, and several other Chicago attractions. If you have a question about directions, please ask at the PHYSOR-2004 Registration desk, or the meeting office located in the Skyway Conference Center, Suite 284.



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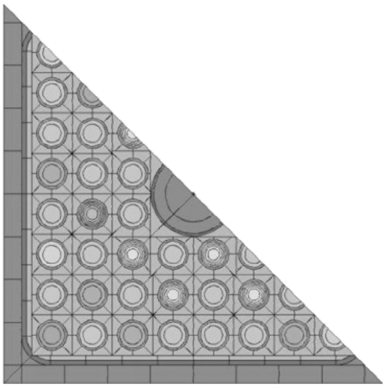
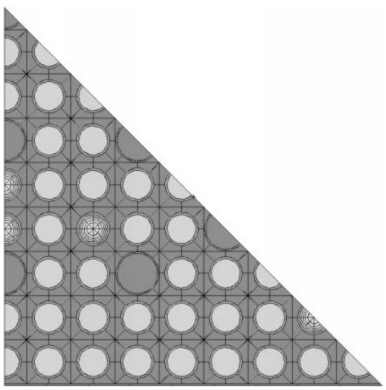
Shiroya, Seiji	Kyoto Univ.
Smith, Michael	ANL
Smith, Nigel	
Stenberg, Chuck	ANL
Stillman, John	ANL
Taiwo, Temitope	ANL
Takeda, Toshikazu	Osaka Univ.
Todosow, Michael	BNL
Uddin, Rizwan	UIUC
Ueki, Taro	
Unesaki, Hironobu	Kyoto Univ. - Japan
Velkov, Kiril	
Waters, Laurie	LANL
Wemple, Charles	INEEL
Yang, Won Sik	ANL
Zaetta, Alain	CEA - France

MCNP Criticality Workshop, April 25, 2004, 1:00 P.M. to 5:00 P.M.





The session will include an overview of MCNP criticality calculations and specific discussions of new features in MCNP5. These topics include random number generation, mesh tallies, the ENDF66 and SAB2002 nuclear data libraries, and generation of temperature-specific nuclear data libraries. In addition, there will be discussion of additional capabilities being developed for the next release of MCNP, such as determination of the dominance ratio, improved stationarity tests, and perturbation improvements. This workshop will be of interest not only to MCNP users but also to anyone who uses Monte Carlo codes for criticality applications. Presenters will be Forrest Brown and Russ Mosteller from the MCNP team at Los Alamos National Laboratory.

This workshop is free to those registered for PHYSOR-2004.

LOCATION AND DATE: Columbus Hall EF, Hyatt Regency Hotel, Chicago, Illinois, U.S.A, April 25, 2004

WORKSHOP ORGANIZERS
Dr. Forrest B. Brown, Monte Carlo Development Team Leader, Los Alamos National Laboratory
Dr. Russell D. Mosteller, Monte Carlo Development Team, Los Alamos National Laboratory

Workshop on Reactor Physics Advances for Design and Analysis of Generation IV Nuclear Energy Systems, April 30, 2004, 8:30 A.M. to 5:00 P.M.

Six nuclear energy systems have been selected for development under the Generation IV nuclear system initiative. The reactor technologies used in these systems are the Very High Temperature Reactor (VHTR), Supercritical Water-Cooled Reactor (SCWR), Gas-Cooled Fast Reactor (GFR), Lead-Cooled Fast Reactor (LFR), Sodium-Cooled Fast Reactor (SFR), and Molten Salt Reactor (MSR). These systems target significant advances over current generation and evolutionary systems in the areas of sustainability (encompassing waste generation and resource utilization), economics, safety and reliability, and proliferation resistance and physical protection. Reactor concept development and core design (including reactor physics) studies are ongoing for these systems in several countries. Analytical tools for evaluating the systems are also being developed. The intent of this Workshop is to have international experts present/discuss ongoing research and development activities in the design and evaluation of the advanced nuclear systems. Core physics design and nuclear data needs and issues will be discussed along with proposals for international collaborations for meeting these needs.

This one-day workshop will have presentations and discussion sessions, and is free to PHYSOR-2004 participants.

LOCATION AND DATE: Columbus Hall GH, Hyatt Regency Hotel, Chicago, Illinois, U.S.A, April 30, 2004

WORKSHOP ORGANIZERS

Dr. H. S. Khalil, Director, Nuclear Engineering Division (NED), ANL and National Technical Director, Design and Evaluation Methods, U.S. Gen IV Program
Dr. R. Jacqmin, Research Director, Nuclear Energy Division, CEA, France.

Dr. T. A. Taiwo, Manager, Nuclear Systems Modeling Section, NED, ANL, U.S.A.

AGENDA

8:30 – 8:45 a.m.	**Introductory Remarks – Hussein S. Khalil (ANL)**
	INTERNATIONAL PERSPECTIVES AND PROPOSALS
8:45 – 9:15 a.m.	**France: Gas-Cooled Reactor Core Physics R&D Activities in France – Robert Jacqmin (CEA)**
9:15 – 9:45 a.m.	**Japan: Recent Development of Fast Reactor Analysis System in Japan – Taira Hazama (JNC)**
9:45 – 10:15 a.m.	**U.S.: Summary of Gen IV Reactor Physics Workshops –Temitope Taiwo (ANL)**
10:15 – 10:30 a.m.	**Break**
10:30 – 11:00 a.m.	**ROK: Gen IV Reactor Physics Developments – Chang Hyo Kim (Seoul Natl. Univ.)**
11:00 – 11:30 a.m.	**EC Involvement in Nuclear Data Needs for Gen IV Systems – Franz-Josef Hambsch (EC-JRC-IRMM)**
11:30 – 1:00 p.m.	**Lunch/Break**
	DISCUSSION AND RECOMMENDATION SESSIONS
1:00 – 2:30 p.m.	**Advanced Reactor Physics Methods and V&V of Physics Tools – lead: David Nigg (INEEL)**
2:30 – 4:00 p.m.	**Nuclear Data Needs and Developments – lead: Gerald Rimpault (CEA)**
4:00 – 5:00 p.m.	**Recommendation and Wrap-up Session – lead: Dan Ingersol (ORNL)**

PHYSOR-2004 Meeting Program

Monday, April 26, 2004, 8:30 A.M.

Plenary Session - International Perspectives in Reactor Physics ***Grand AB***
Session Chair: Massimo Salvatores (ANL/CEA)

Preparing the Future: New Challenges for Nuclear Energy Systems, Jacques Bouchard (French Atomic Energy Commission - CEA)

Strengthening of Physical Base of Nuclear Power, Sergei. M. Zaritsky (Kurchatov Institute)

Innovation Dynamics and Nuclear Power, Yoshiaki Oka (University of Tokyo)

The 21st Century Rise of Nuclear Power, Burton Richter (Stanford Linear Accelerator Center)

Monday, April 26, 2004, 12:00 P.M.

Luncheon ***Columbus IJKL***

Historical Perspectives on Reactor Physics Experiments, Massimo Salvatores (ANL/CEA)

Monday, April 26, 2004, 1:30 P.M.

Session 1A International Collaboration in Reactor Physics ***Columbus EF***

Session Organizer: Thomas J. Downar (Purdue University).
Session Chairs: Thomas J. Downar (Purdue University), Jean-Pascal Hudelot (CEA Cadarache).

1:30 PM	The Numerical Nuclear Reactor for High Fidelity Integrated Simulation of Neutronic, Thermal-Hydraulic and Thermo Mechanical Phenomena - Project Overview, David P. Weber, T. Sofu, P. A. Pfeiffer, W. S. Yang, T. A. Taiwo (ANL), H. G. Joo, J. Y. Cho, K. S. Kim, T. H. Chun (KAERI), T. J. Downar, J. W. Thomas, Z. Zhong (Purdue Univ.), C. H. Kim, B. S. Han (Seoul National Univ.)
1:50 PM	Methods and Performance of a Three-Dimensional Whole-Core Transport Code DeCART, Han Gyu Joo, Jin Young Cho, Kang Seog Kim, Chung Chan Lee, Sung Quun Zee (Korea Atomic Energy Research Institute)
2:10 PM	Consistent Comparison of Monte Carlo and Whole-Core Transport Solutions for Cores with Thermal Feedback, Han Gyu Joo, Jin Young Cho, Kyo Youn Kim, Moon Hee Chang (Korea Atomic Energy Research Institute), Beom Seok Han, Chang Hyo Kim (Seoul National University)
2:30 PM	Evaluation of Turbulence Models for Flow and Heat Transfer in Fuel Rod Bundle Geometries, Tanju Sofu (ANL), T. H. Chun, W. K. In (KAERI)
2:50 PM	Methodology for Coupling Computational Fluid Dynamics with Integral Transport Neutronics, Justin W. Thomas, Zhaopeng Zhong (Purdue University), Tanju Sofu (ANL), Thomas J. Downar (Purdue University)
3:10 PM	*Break*
3:30 PM	Coupled Calculations using the Numerical Nuclear Reactor for Integrated Simulation of Neutronic and Thermal-Hydraulic Phenomena, David P. Weber, Tanju Sofu, Won Sik Yang (ANL), Thomas J. Downar, Justin W. Thomas, Zhaopeng Zhong (Purdue University), Han Gyu Joo (KAERI)
3:50 PM	OSMOSE: An Experimental Program for the Qualification of Integral Cross Sections of Actinides, Jean-Pascal Hudelot (CEA Cadarache), Raymond Klann (ANL), Phillippe Fougeras (CEA Cadarache), Xavier Genin, Nicolas Drin (CEA Valrho), Louis Donnet (CEA)
4:10 PM	MINERVE Reactor Characterization in Support of the OSMOSE Program: Spectral Indices, Raymond Klann (ANL), Jean-Pascal Hudelot, Muriel Antony (CEA Cadarache), Bradley Micklich, George Imel (ANL), Gregory Perret (CEA), Jean-Michel Girard, Valerie Laval (CEA Cadarache)
4:30 PM	HTR-N Plutonium Cell Burnup Benchmark: Definition, Results & Intercomparison, Jim C. Kuijper (NRG), N. Cerullo (Universita di Pisa), F. Damian (CEA), J.L. Kloosterman (Delft University of Technology), G. Lomonaco (Universita di Pisa), J. Oppe (NRG), X. Raepsaet (CEA), H.J. Ruetten (FZJ)

Session 1B Monte-Carlo Methods and Developments ***Columbus G***

Session Organizers: Forrest Brown (LANL), Richard Sanchez (CEA), Laurie Waters (LANL).
Session Chairs: J. Eduard Hoogenboom (Delft University of Technology), William R. Martin (University of Michigan).

1:30 PM	Monte Carlo Parameter Studies and Uncertainty Analyses with MCNP5, Forrest Brown, Jeremy E. Sweezy (LANL), Robert Bruce Hayes (Washington TRU Solutions)
1:50 PM	Theoretical and Practical Study of the Variance and Efficiency of a Monte Carlo Calculation due to Russian Roulette, J. Eduard Hoogenboom (Delft University of Technology)
2:10 PM	Variance Reduction Techniques for the Monte Carlo Simulation of Neutron Noise Measurements, Máté Szieberth (Budapest University of Technology and Economics), Jan Leen Kloosterman (Delft University of Technology)
2:30 PM	Two Dimensional Functional Expansion Tallies for Monte Carlo Simulations, David P. Griesheimer, William R. Martin (University of Michigan)
2:50 PM	Point KENO V.a: A Continuous-Energy Monte Carlo Code for Criticality Safety Applications, Michael Dunn, Maurice Greene, Dan F. Hollenbach, L. M. Petrie (ORNL)
3:10 PM	*Break*
3:30 PM	Eigenfunction Convergence and Transmutation Enhancements in MCNPX, Gregg McKinney, Holly Trellue, John Hendricks, Laurie Waters (LANL)
3:50 PM	Continuously Varying Material Properties and Tallies for Monte Carlo Calculations, Forrest Brown (LANL), David P. Griesheimer, William R. Martin (University of Michigan)
4:10 PM	Calculation of the Effective Delayed Neutron Fraction Using Monte Carlo Techniques, Steven C. Van Der Marck, Robin Klein Meulekamp (NRG)
4:30 PM	Benchmark of MONTEBURNS against Measurements on Irradiated UOX and MOX Fuels, Christos Trakas, Lucien Daudin (FRAMATOME-ANP)

Session 1C Reactor Physics Benchmarks and Experiments ***Columbus H***

Session Organizers: Russell D. Mosteller (LANL), Hironobu Unesaki (Kyoto University).
Session Chairs: Rakesh Chawla (Paul Scherrer Institute), Jess Gehin (ORNL).

1:30 PM	Analysis of Benchmark Results for Reactor Physics of LWR Next Generation Fuels, Takanori Kitada (Osaka University), Keisuke Okumura (Japan Atomic Energy Research Institute), Hironobu Unesaki (Kyoto University), Etsuro Saji (Secretariat of The Nuclear Safety Commission)
1:50 PM	Improvements of Isotopic Ratios Prediction through TAKAHAMA-3 Chemical Assays with the JEFF3.0 Nuclear Data Library, Arnaud Courcelle, Alain Santamarina, Stéphane Mengelle (CEA)
2:10 PM	Analysis of the HTR-10 Initial Core with a Monte Carlo Code MVP, Yasunobu Nagaya, Keisuke Okumura, Takamasa Mori (JAERI), Wataru Nakazato (Tokyo Institute of Technology)
2:30 PM	OECD/NEA International Benchmark on 3-D VENUS-2 MOX Core Measurements, Nadia Messaoudi (Belgian Nuclear Research Centre, SCK·CEN), Byung-chan Na (OECD/Nuclear Energy Agency)
2:50 PM	BN-600 Full MOX Core Benchmark Analysis, Y. I. Kim (IAEA), R. Hill, K. Grimm (ANL), G. Rimpault (CEA Cadarache), T. Newton (Serco Assurance), Z. H. Li (CIAE), A. Rineiski (FZK/IKET), P. Mohanakrishnan (IGCAR), M. Ishikawa (Japan Nuclear Cycle Development Institute), K. B. Lee (KAERI), A. Danilytchev, V. Stogov (IPPE)
3:10 PM	*Break*
3:30 PM	JOYO MK-III Performance Test at Low Power and Its Analysis, Gou Chiba, Kenji Yokoyama, Shigetaka Maeda, Takashi Sekine (Japan Nuclear Cycle Development Institute)
3:50 PM	High Moderation BWRs Fully Loaded with MOX Fuel: The BASALA Experimental Programme, Stephane Cathalau, Patrick Blaise, Philippe Fougeras, Nicolas Thiollay, Alain Santamarina, Olivier Litaize (CEA), Toru Yamamoto, Ryoji Kanda, Masaru Sasagawa, Takuya Umano, Tsukasa Kikuchi (NUPEC), Jean-Louis Nigon (COGEMA)
4:10 PM	The Experimental Determination of the Relative Abundances and Decay Constants of Delayed Neutrons of the IPEN/MB-01 Reactor, Adimir Dos Santos, Ricardo Diniz, Rogerio Jerez, Luis Antonio Mai, Mitsuo Yamaguchi Graciete S. A/ Silva (Instituto de Pesquisas Energeticas e Nucleares), Arlindo Gilson Mendonca (Centro Technologico da Marinha em Sao Paulo)
4:30 PM	The Experimental Determination of the Effective Delayed Neutron Parameters: B_{eff}, B_{eff}/Λ and Λ of the IPEN/MB-01 Reactor, Adimir Dos Santos, Ricardo Diniz, Leda C. C. B. Fanaro, Rogerio Jerez, Graciete S. A. Silva, Mitsuo Yamaguchi (Instituto de Pesquisas Energeticas e Nucleares)

Session 1D Reactor Analysis Methods ***Grand B***

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chairs: Giovanni Bruna (Framatome ANP), Giuseppe Palmiotti (ANL).

1:30 PM	Status of Reactor Analysis Methods and Codes in the U.S.A., Won Sik Yang, Temitope A. Taiwo (ANL)
1:50 PM	Validation of WIMS9, Tim Newton, Les Hutton, Ray Perry (Serco Assurance)
2:10 PM	Application of the DSA Preconditioned GMRES Formalism to the Method of Characteristics - First Results , Romain Le Tellier, Alain Hebert (Ecole Polytechnique de Montreal)
2:30 PM	Improvement of the SPH Method for Multi-Assembly Calculations, Akio Yamamoto, Masahiro Tatsumi, Yasunori Kitamura, Yoshihiro Yamane (Nagoya University)
2:50 PM	Method of Characteristics Applied to a MTR Whole Core Modeling, Alain Aggery, Corinne D'aletto, Jacques Di-Salvo, Richard Sanchez, Simone Santandrea, Michel Soldevila, Guy Willermoz (CEA)
3:10 PM	*Break*
3:30 PM	3D Characteristics Method with Linearly Anisotropic Scattering, Mohamed Dahmani, Robert Roy, Jean Koclas (École Polytechnique de Montréal)
3:50 PM	A Mutual Resonance Shielding Model Consistent with Ribon Subgroup Equations, Alain Hebert (Ecole Polytechnique de Montreal)
4:10 PM	Sensitivity Studies on Cross-Section Generation and Modeling for BWR Core Simulation Using SAPHYR Code System, Nadejda Todorova, Kostadin Ivanov (Pennsylvania State University), Emmanuel Rigaut, Eric Denis Royer (CEA)
4:30 PM	About Calculation of Axial Diffusion Coefficient in Nuclear Reactor Cells, Tamara Semeonovna Poveschenko, Nickolay Ilyich Laletin (RRC 'Kurchatov Institute')

Monday, April 26, 2004, 5:00 P.M.

Special Session - Reactor Physics for Future Nuclear Systems ***Grand AB***
Session Chair: Kevan Weaver (INEEL)

Advances in Reactor Physics Analysis Capabilities Needed for Design of Generation IV Systems, Hussein Khalil (Argonne National Laboratory)

*Very High Temperature Reactor: Promising Nuclear System, Specific Reactor Physic*s, Jean-Claude Gauthier (Framatome)

Tuesday, April 27, 2004, 8:30 A.M.

Plenary Session - Current Needs in Reactor Physics ***Grand AB***
Session Chair: Yoshiaki Oka (University of Tokyo)

Needs Related to the Nuclear Power Industry, Kord Smith (Studsvik-Scandpower)

Needs Related to Nuclear Fuel Cycle Initiatives, Christine Brown (British Nuclear Fuels plc)

Tuesday, April 27, 2004, 10:00 A.M.

Session 2A Deep-Burn Physics and Methodology ***Columbus EF***

Session Organizers: Alan Baxter (General Atomics), Giovanni Bruna (Framatome ANP), Alexander Stanceulescu (IAEA).
Session Chair: Giovanni Bruna (Framatome ANP).

10:00 AM	Uncertainty Analysis and Optimization Studies on the Deep-Burner Modular Helium Reactor (DB-MHR) for Actinide Incineration, Giovanni B. Bruna, Rocco Labella, and Christos Trakas (FRAMATOME-ANP), Alan Baxter and Carmelo Rodriguez (General Atomics), Francesco Venneri (LANL)
10:20 AM	PBMR Deep-Burn: A Pebble-Bed HTGR Burning Its Own 'Waste', Dirceu F. Da Cruz, J.B.M. De Haas, A.I. Van Heek (NRG)
10:40 AM	HTGR Actinide Burner Feasibility Studies: Calculation Scheme Related Considerations, Frederic Damian (CEA), Patrick Blanc-Tranchant (CEA - Saclay), Xavier Raepsaet (CEA - Saclay), Jean-Christophe Klein (CEA - Cadarache), Florence Dolci, Oliver Koberl (CEA)

11:00 AM	Studies of a Deep Burn Fuel Cycle for the Incineration of Military Plutonium in the GT-MHR using the Monte-Carlo Burnup Code, Alberto Talamo, Waclaw Gudowski (Royal Institute of Technology)
11:20 AM	Spectral Shift Methodology for 'Deep-Burnup' of Uranium-Thorium-Hydride-Fuel, Zeev Shayer (University of Denver)
11:40 AM	Cascade Reactor Concept for Neutron Multiplication of Subcritical Core, Kazumi Ikeda, Takako Shiraki (Mitsubishi Heavy Industries, Ltd.)

Session 2B Accelerator Applications and Spallation Physics ***Columbus G***

Session Organizers: Eric Pitcher (LANL), Stefano Monti (ENEA), Hiroyuki Oigawa (JAERI).
Session Chair: Eric Pitcher (LANL).

10:00 AM	Research and Development Activities for Accelerator Driven System at JAERI, Kazufumi Tsujimoto, Toshinobu Sasa, Kenji Nishihara, Hiroyuki Oigawa, Hideki Takano (Japan Atomic Energy Research Institute)
10:20 AM	Research on Accelerator Driven Subcritical Reactor at Kyoto University Critical Assembly (KUCA) with the FFAG Proton Accelerator, Tsuyoshi Misawa, Hironobu Unesaki, Cheol Ho Pyeon, Chihiro Ichihara (Kyoto University), Yasunori Kitamura (Nagoya University) and Seiji Shiroya (Kyoto University)
10:40 AM	The TRADE Experiment: Status of the Project and Physics of the Spallation Target, Stefano Monti, Carlo Rubbia (ENEA), Massimo Salvatores (CEA/ANL), Nunzio Burgio, Mario Carta, Pietro Agostini, Fabrizio Pisacane, Alfonso Santagata (ENEA), Cecille Krakowiak-Aillaud (CEA), Yacine Kadi (European Organization for Nuclear Research), Eric Pitcher (LANL), Jean-Claude Steckmeyer (CNRS), Cornelis Broeders, Dankward Struwe (Karlsruhe), Adonai Herrera-Martinez (University of Cambridge)
11:00 AM	Impact of Heterogeneous Cm Distribution on Proton Source Efficiency in Accelerator-Driven Systems, Per Seltborg, Jan Wallenius, Waclaw Gudowski (Royal Institute of Technology)
11:20 AM	Reactor-Accelerator Coupled Experiments (RACE): A Feasibility Study at TAMU, William S. Charlton, Venkat Krishna Taraknath Woddi (Texas A&M University)
11:40 AM	Towards an Improved GELINA Neutron Target, Marek Flaska, Arjan Plompen, Willy Mondelaers (IRMM-JRC-EC), Danny Lathouwers, Tim Van Der Hagen Jr., Hugo Van Dam (Delft University of Technology)

Session 2C Reactor Physics and Materials Issues ***Columbus H***

Session Organizer: Abderrafi M. Ougouag (INEEL). Session Chair: Abderrafi M. Ougouag (INEEL).

10:00 AM	Displacement Kerma Cross Sections for Neutron Interactions in Molybdenum, Abderrafi Ougouag, Charles A. Wemple, Clinton Van Siclen (INEEL)
10:20 AM	Thermal Neutron Scattering Cross Sections of Thorium Hydride, Iyad I. Al-qasir, Ayman I. Hawari, Victor Hugo Gillette, Bernard W. Wehring, Tong Zhou (North Carolina State University)
10:40 AM	Three-Dimensional RAMA Fluence Methodology Benchmarking, Steven Baker (TransWare Enterprises Inc.), Robert G. Carter (EPRI), Kenneth E. Watkins, Dean B. Jones (TransWare Enterprises Inc.)
11:00 AM	Ab Initio Generation of Thermal Neutron Scattering Cross Sections, Ayman I. Hawari, Iyad I. Al-qasir, Victor Hugo Gillette, Bernard W. Wehring, Tong Zhou (North Carolina State University)
11:20 AM	Support Vector Machine in Classification of Positron Lifetime Spectra, Senada D. Avdic (University of Tuzla, Boznia)
11:40 AM	Research Reactor Application to Iridium-192 Production for Cancer Treatment, Maria Elisa C. M. Rostelato, Constancia Silva, Paulo Roberto Rela, Carlos Zeituni, Vladimir Lepki, Anselmo Feher (Nuclear Energy National Commission, CNEN - BRAZIL)

Session 2D Advances in LWR Analyses ***Grand B***

Session Organizer: Marvin Adams (Texas A&M University).
Session Chairs: Marvin Adams (Texas A&M University), Dmitriy Anistratov (North Carolina State University)

10:00 AM	Renormalized Treatment of the Double Heterogeneity with the Method of Characteristics, Richard Sanchez (CEA)
10:20 AM	Coarse-Mesh Discretized Low-Order Quasidiffusion Equations for Subregion Averaged Scalar Fluxes, Dmitriy Y. Anistratov (North Carolina State University)
10:40 AM	Application of a Heterogeneous Coarse Mesh Transport Method to a MOX Benchmark Problem, Benoit Forget, Farzad Rahnema, Scott W. Mosher (Georgia Institute of Technology)
11:00 AM	Low-Order Quasidiffusion Equations in 2D Geometry, Hikaru Hiruta, Dmitriy Y Anistratov (North Carolina State University)
11:20 AM	The Use of an Artificial Neural Network for On-Line Prediction of Pin-Cell Discontinuity Factors in PARCS, Tomasz Kozlowski, Deokjung Lee, Thomas J. Downar (Purdue University)

11:40 AM	Developing a Basis for Predicting and Assessing Trends in BWR Core Tracking, Anna Smolinska, John Rea, Atul Karve, Kenneth Gardner (Global Nuclear Fuel)

Tuesday, April 27, 2004, 1:00 P.M.

Plenary Session - Experimental Activities in Reactor Physics ***Grand AB***
Session Chair: Sergei. M. Zaritsky (Kurchatov Institute)

The International Program TRADE - TRIGA Accelerator Driven Experiments, George Imel (Argonne National Laboratory)

Nuclear Cross Section Measurements within the Advanced Fuel Cycle Initiative, Eric Pitcher (Los Alamos National Laboratory)

Tuesday, April 27, 2004, 2:30 P.M.

Session 3A Reactor and Neutron Physics ***Columbus EF***

Session Organizers: Enrique M. Gonzalez-Romero (CEEMT), Robert N. Hill (ANL).
Session Chairs: Enrique M. Gonzalez-Romero (CEEMT), Robert N. Hill (ANL).

2:30 PM	New Resonant Mixture Self-Shielding Treatment in the APOLLO2 Code, Mireille Coste-Delclaux, Stephane Mengelle (CEA)
2:50 PM	Extension of KIN3D, a Kinetics Capability of VARIANT, for Modeling Fast Transients in Accelerator Driven Systems, Cristian Rabiti, Andrei Rineiski (FZK/IKET)
3:10 PM	Reactivity Assessment and Spatial Time-Effects from the MUSE Kinetics Experiments, Mario Carta, Antonio D'angelo, Vincenzo Peluso (ENEA), Gerardo Aliberti, Giuseppe Palmiotti, George Imel (ANL), Jean Francois Lebrat (CEA), Enrique M. Gonzalez-Romero (Centro de Estudios Energéticos Medioambientales y Tecnológicos), David Villamarin (CIEMAT), Sandra Dulla , Fabrizio Gabrielli, Piero Ravetto (Politecnico di Torino), Massimo Salvatores (ANL)
3:30 PM	Monte Carlo Modeling of a Time-of-Flight (ToF) Experiment for Determination of Fe Scattering Cross Sections, Michael T. Wenner, Alireza Haghighat (University of Florida), James M. Adams, Allan D. Carlson (National Institute of Standards and Technology), Steven M. Grimes, Thomas N. Massey (Ohio University)
3:50 PM	Application of the Dynamic Control Rod Reactivity Measurement Method to Korea Standard Nuclear Power Plants, Eun-ki Lee, Ho-chul Shin, Suk-Jin Ryu, Sung-Man Bae, Yong-Kwan Lee (Korea Electric Power Research Institute - KEPRI)
4:10 PM	Physics Characteristics of U-ZrH1.6 Fueled PWR Cores, Zeev Shayer, Ehud Greenspan (University of California)
4:30 PM	CANDU Adjuster Rods Incremental Cross Sections Evaluation: A Perturbative Approach, Guy Marleau (Ecole Polytechnique de Montreal)

Session 3B Research Reactors ***Columbus G***

Session Organizers: Nelson Hanan (ANL), Hamid Ait Abderrahim (SCK-CEN).
Session Chairs: R. Trenton Primm III (ORNL), Thomas H. Newton Jr. (MIT).

2:30 PM	Modeling the MIT Reactor Neutronics for LEU Conversion Studies, Thomas H. Newton Jr., Zhiwen Xu, Edward E. Pilat, Mujid Kazimi (Massachusetts Institute of Technology)
2:50 PM	A New De-Homogenisation Method for Local Power Reconstruction, Guy Pierre Willermoz, David Blanchet, Jacques Di-Salvo, Christoph Doderlein, Nicolas Huot (CEA)
3:10 PM	Reactor Physics Studies of Reducted-Tantalum-Content Control and Safety Elements for the High Flux Isotope Reactor, Trent Primm (ORNL)
3:30 PM	Automated Three Dimensional Depletion Capability for the Pennsylvania State University Research Reactor, Chanatip Tippayakul, Kostadin Ivanov, C.F. Sears, G.M. Morlang, B.J. Heidrich (Pennsylvania State University)
3:50 PM	HOR: Criticality Comparison Using a Nodal Code, Monte Carlo Codes and Plant Data, P.F.A. de Leege (Delft University of Technology), Frederik Reitsma (NECSA)
4:10 PM	The Application of the Zr to the Thermal Neutron Fluence Monitoring at the Irradiation Experiments of the Research Reactor, Myong Seop Kim, Sang Jun Park, Byung Chul Lee, Heonil Kim, Byung Jin Jun (Korea Atomic Energy Research Institute)

4:30 PM Determination of the Linear Power in MOX Fuel Rods Irradiated at the BR2 Reactor, V. Kuzminov, M. Weber, E. Koonen, (SCK-CEN (Belgian Nuclear Research Center))

Session 3C Fuel Cycle Physics ***Columbus H***

Session Organizers: Alain Zaetta (CEA Cadarache), Michael Todosow (BNL).
Session Chairs: Michael Todosow (BNL), John Stillman (ANL).

2:30 PM The Physics of TRU Transmutation -- A Systematic Approach to the Intercomparison of Systems, Massimo Salvatores (CEA & ANL), Robert N. Hill (ANL), Igor Slessarev, Gilles Youinou (CEA Cadarache)

2:50 PM Dynamic Analysis of the AFCI Scenarios, Abdellatif M. Yacout, Robert N. Hill, Luc Van Den Durpel, Phillip J. Finck (ANL), Erich Schneider, Charles G. Bathke (LANL), J. S. Herring (INEEL)

3:10 PM Uncertainty Analysis on Back-end Fuel Cycle Main Parameters, Giovanni B. Bruna and Bertrand Carlier (FRAMATOME-ANP), Enrico Padovani and Alberto Michelotti (POLITECNICO di MILANO)

3:30 PM Effectiveness of Different Burnable Poisons in a Long Cycle BWR, Yuichiro Inoue, Zhiwen Xu, Edward E. Pilat (Massachusetts Institute of Technology)

3:50 PM Studies of Advanced Fuel Cycles in Indian PHWRs and AHWR, P. D. Krishnani, R. Srivenkatesan, Baltej Singh, Umasankari Kannan, Arvind Kumar, Sadhana Mukerji (Bhabha Atomic Research Center)

4:10 PM Influence of Nuclear Fuel Cycle Duration and Reprocessing Losses Level on the Nuclear Power System Structure, Stanislav A. Subbotin, Pavel N. Alekseev, Anatoly A. Dudnikov (Russian Research Center Kurchatov Institute)

4:30 PM Assessment of Reduced Moderation Water Reactor Fuel Cycle, Taek Kyum Kim, Won Sik Yang, Temitope A. Taiwo, Robert N. Hill (ANL)

Session 3D Criticality Benchmarks ***Grand B***

Session Organizers: Russell D. Mosteller (LANL), Hironobu Unesaki (Kyoto University).
Session Chairs: Richard D. McKnight (ANL), Hironobu Unesaki (Kyoto University).

2:30 PM TransLAT Lattice Physics Code Benchmark to the B&W Gadolinia Criticals, Steven Baker (TransWare Enterprises Inc.)

2:50 PM ENDF/B-V and ENDF/B-VI Calculations for the LWBR SB Core Benchmarks with MCNP5, Russell D. Mosteller (LANL)

3:10 PM Benchmark Comparisons of Deterministic and Monte Carlo Codes for a PWR Heterogeneous Assembly Design, Taek K. Kim, John Stillman, Temitope A. Taiwo (ANL), Christine Chabert, Laurence Mandard (CEA)

3:30 PM Analysis of the Experimental Program MISTRAL Using CASMO-4, Akiko Kanda, Tetsuo Nakajima, Kyoko Sano (Japan Nuclear Energy Safety Organization)

3:50 PM Solution of the C5G7 3-D Extension Benchmark by the SN Code TORT, Armin Seubert, Winfried Zwermann, Siegfried Langenbuch (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH)

4:10 PM Tort Solutions to the Three-Dimensional MOX Neutron Transport Benchmark Problem, 3-D Extension C5G7MOX, Jesse Klingensmith, Yousry Youssef Azmy (Pennsylvania State University), Jess Gehin (ORNL), Roberto Orsi (ENEA FIS-NUC)

4:30 PM Creation of a Simplified Benchmark Model for the Neptunium Sphere Experiment, Russell D. Mosteller, David J. Loaiza, Rene G. Sanchez (LANL)

Tuesday, April 27, 2004, 5:00 P.M.

Session 4A Poster Session and Reception ***Crystal Ballroom Foyer***

Monte-Carlo Techniques to Simulate Pebble Dislocations in a PB-MR during Depletion, Nunzio Burgio (ENEA - CRE Casaccia), Giovanni Bruna (Framatome), Alfonso Santagata (ENEA - CRE Casaccia), Rocco Labella, Engineer (FRAMATOME-ANP), Jessica Beati, Augusto Gandini (University of Rome La Sapienza)

Design of a Very High Temperature Pebble-Bed Reactor Using Genetic Algorithms, Hans D. Gougar, Abderrafi M. Ougouag, William K. Terry (INEEL), Kostadin Ivanov (Pennsylvania State University)

Feasibility of Using Burnable Poisons for Reduction of Coolant Void Reactivity in LMR for TRU Transmutation, Yong Nam Kim, Jong Kyung Kim (Hanyang University), Won Seok Park (Korea Atomic Energy Research Institute)

Plutonium Disposition in the PBMR-400 High-Temperature Gas-Cooled Reactor, Eben J. Mulder, Eberhard Teuchert (PBMR (Pty) Ltd)

Definition of a Calculation Scheme with MONTEBURNS for Decay Heat Calculations of HTR Fuel, Anne De L'hermite (FRAMATOME-ANP)

Concept of a Gas Cooled Fast Reactor, N. Kodochigov, N. Kuzavkov, Yu. Sukharev, S. Usynia (OKB Mechanical Engineering)

MINERVE Reactor Characterization in Support of the OSMOSE Program: Safety Parameters, Jean-Pascal Hudelot (CEA Cadarache), Raymond Klann, Bradley Micklich (ANL), Muriel Antony, Pierre Leconte, Jean-Michel Girard, Valerie Laval (CEA - Cadarache), Gregory Perret (CEA)

Thermo Mechanical Calculations for Integrated High Fidelity Reactor Core Simulations, Phillip A. Pfeiffer, Tanju Sofu (ANL), Z. Dong (University of Chicago)

Monte Carlo Midway Forward-Adjoint Coupling with Legendre Polynomials for Borehole Logging Applications, David Legrady, J. Eduard Hoogenboom (Delft University of Technology)

Visualization of Space-Dependency of Responses of Monte Carlo Calculations Using Legendre Polynomials, J. Eduard Hoogenboom, David Legrady (Delft University of Technology)

An Influence of Core Physics Peculiarities upon the Thermal Hydraulics Performance in Cascade Subcritical Molten Salt Reactor, Vladimir A. Nevinitsa, Pavel N. Alekseev, Aleksey A. Sedov, Peter A. Fomichenko, Alexander V. Vasiliev, Elyaz M. Aisen (Russian Research Center Kurchatov Institute), Andrey M. Voloshchenko (Keldysh Institute of Applied Mathematics)

AGENT Code: Open-Architecture Analysis and Configuration of Research Reactors – Graphic Tools, Tatjana Jevremovic, Dimitrios Andristos (Purdue University)

DELPHI:A New Subcritical Assembly at Delft University of Technology, Jan Leen Kloosterman (Delft University of Technology)

The Studies of RBMK-1500 Reactor Core Behavior during Abnormal Operation Transients, Arvydas Adomavicius, Anton Belousov, Jonas Gylys, Viktor Ognerubov, Stanislovas Ziedelis (Kaunas University of Technology), Audrius Jasiulevicius, Andrej Kubarev (Royal Institute of Technology)

Onboard Radiation Shielding Estimates for Interplanetary Manned Missions, Aaron Totemeier, Tatjana Jevremovic, Derek Hounshel (Purdue University)

Reduction of Cross-Section-Induced Errors of the BN-600 Hybrid Core Nuclear Parameters by Using BFS-62 Critical Experiment Data, Akira Shono, Taira Hazama, Makoto Ishikawa (Japan Nuclear Cycle Development Institute), Gennadi Manturov (Institute of Physics & Power Engineering)

Uncertainty Analysis Results on BN-600 Hybrid Core Nuclear Physics Characteristics, Gennadi Manturov, Mikhail Semenov, Anatoly Seregin, Anatoly Tsiboulya (Institute of Physics and Power Engineering - IPPE), Taira Hazama (Japan Nuclear Cycle Development Institute)

Analysis of SNEAK-7A & 7B Critical Benchmarks using3-D Deterministic Transport and Sensitivity, Sang Kim (KAERI), Ivo Kodeli, Enrico Sartori (OECD Nuclear Energy Agency)

Derivation of the Space and Energy Dependent Formula for the Third Order Neutron Correlation Technique, Tomohiro Endo, Yasunori Kitamura, Akio Yamamoto, Yoshihiro Yamane (Nagoya University)

Cross-Section Generation for TRADE Fuel, Ron Dagan, Cornelis H. M. Broeders, Madelena Badea, Anton Travleev (Forschungzentrum Karlsruhe(FZK))

On-Line Determination of the Prompt Fraction of In-Core Neutron Detectors in CANDU Reactors, Christophe Demazière, Oszvald Glockler (Ontario Power Generation)

Dynamics of a Reduced Order Model of Natural Circulation BWR, Quan Zhou, Rizwan Uddin (University of Illinois at Urbana-Champaign)

Solution of the 1D Kinetic Diffusion Equations Using a Reduced Nodal Cubic Scheme, Edmundo Del Valle (Instituto Politecnico Nacional), Armando Gomez, Gustavo Alonso, Arturo Delfin (Instituto Nacional de Investigaciones Nucleares)

Implementation of a Full P1 Method in the Diffusion Code DONJON/NDF, Jean Koclas (École Polytechnique de Montréal), Benoit Forget (Georgia Institute of Technology)

The Simplified Even-Parity SN Equations Applied to a MOX Fuel Assembly Benchmark Problem on Distributed Memory Environments, Gianluca Longoni, Alireza Haghighat (University of Florida)

On the Influence of Differences Between Various Group Microconstant Libraries and Between Different Transport Options on Calculation Results for Cells and Subassemblies of VVER-1000 Reactor, Nickolay Ilyich Laletin, Nikolay Sultanov Sr., Aleksey Kovalishin Sr. (RRC Kurchatov Institute)

Sipping Tests on a Failed Irradiated MTR Fuel Element, Carlos Alberto Zeituni, Luis Antonio Albiac Terremoto, José Eduardo Rosa Da Silva (Instituto de Pesquisas Energéticas e Nucleares)

Capturing the Effects of Unlike Neighbors in Single-Assembly Caculations, Kevin Clarno, Marvin Adams (Texas AM University).

Tuesday, April 27, 2004, 7:00 P.M.

Banquet — ***Crystal Ballroom***

Keynote Speaker: Alan Waltar (Pacific Northwest National Laboratory)

Wednesday, April 28, 2004, 8:30 A.M.

Session 5A Fuel/Core Design and Analysis — ***Columbus EF***

Session Organizer: Kord Smith (Studsvik-Scandpower). Session Chair: Richard J. Cacciapouti (AREVA).

8:30 AM	New Computational Methodology for Large 3D Neutron Transport Problems, Mohamed Dahmani, Robert Roy, Jean Koclas (École Polytechnique de Montréal)
8:50 AM	Spent Nuclear Fuel Analyses Based on In-Core Fuel Management Calculations, Sigurd Borresen (Studsvik Scandpower)
9:10 AM	ALAADIN/FLS - A BWR Fast Lattice Design Simulation Tool, Albert G. Gu, Robert J. Veklotz, Hoju Moon, Ralph G. Grummer, Craig Brown (Framatome ANP)
9:30 AM	The Feasibility Study of the Minimum-Shuffling Reloading Strategy for PWR, Masato Tabuchi (Nagoya University), Yasushi Hanayama, Masatoshi Yamasaki (Nuclear Fuel Industries, Ltd.), Akio Yamamoto (Nagoya University)
9:50 AM	Probability Approaching Method (PAM) and Its Application on Fuel Management Optimization, Zhihong Liu, Yongming Hu, Gong Shi (Tsinghua University)
10:10 AM	*Break*
10:30 AM	High Fuel Burn-Up and Nonproliferation in PWR-Type Reactor on the Basis of Modified Th-fuel, Gennady Genrikhovich Kulikov (International Science and Technology Center), Anatoly Nikolaevich Shmelev, Vladimir Alexandrovich Apse (Moscow Engineering Physics Institute)
10:50 AM	Over-Moderated MOX Fuel Assembly in a BWR Mixed Reload, Jose Ramon Ramirez-Sanchez (Instituto Nacional de Investigaciones Nucleares)
11:10 AM	Performance Comparison of Different Absorbent Materials in BWR Control Rods, José Luis Montes Tadeo, Juan J. Ortiz, Raul Perusquia (Mexican Nuclear Research Institute), Robert T. Perry (LANL)
11:30 AM	Enriched Gadolinium as Burnable Absorber for PWR, Klaes-Håkan Bejmer (Vattenfall Bränsle AB, Sweden), Ola Seveborn (Uppsala University)

Session 5B Gas-Cooled Reactors — ***Columbus G***

Session Organizer: Alan Baxter (General Atomics). Session Chair: Alan Baxter (General Atomics).

8:30 AM	The Pebble Bed Modular Reactor Layout and Neutronics Design of the Equilibrium Cycle, Frederik Reitsma (PBMR Pty Ltd)
8:50 AM	Neutronic Modeling for a Gas-Cooled Fast Reactor Assuming Coated Fuel Particles, Hervé Golfier, Laurent Buiron, Christine Poinot, Baptiste Pothet, Jean-François Salavy, Etienne Studer (CEA)
9:10 AM	Methodology for a Large Gas-Cooled Fast Reactor Core Design and Associated Neutronic Uncertainties, Jean-Christophe Bosq, Alain Conti, Gerald Rimpault, Jean-Claude Garnier (CEA)
9:30 AM	Optimal Moderation in the Pebble-Bed Reactor for Enhanced Passive Safety and Improved Fuel Utilization, Abderrafi M. Ougouag, Hans D. Gougar, William K. Terry (Idaho National Engineering and Environmental Laboratory), Ramatsemela Mphahlele, Kostadin Ivanov (Pennsylvania State University)
9:50 AM	Fuel Design and Core Layout for a Gas Cooled Fast Reactor, Wilfred Van Rooijen (Delft University of Technology - Interfaculty Reactor Institute), Jan Leen Kloosterman, Tim Van Der Hagen, Hugo Van Dam (Delft University of Technology)
10:10 AM	*Break*
10:30 AM	GT-MHR Core Modelling: From Reference Modelling Definition in Monte-Carlo Code to Calculation Scheme Validation, Frederic Damian, Xavier Raepsaet, Simone Santandrea, Alain Mazzolo, Christine Poinot (CEA - Saclay), Jean-Christophe Klein (CEA - Cadarache), Leandre Brault, Christian Garat (Framatome - ANP)

10:50 AM Modeling of HTRs with Monte Carlo: Sensitivity due to Different Isotopic Fuel Composition, Rita Plukiene (Physics Institute - Lithuania), Danas Ridikas (CEA Saclay), Arturas Plukis, Vidmantas Remeikis (Physics Institute - Lithuania)

11:10 AM Low-Conversion Ratio Gas-Cooled Fast Reactors, Edward A. Hoffman, Temitope A. Taiwo, Robert N. Hill (ANL)

11:30 AM Possibility to Use Different Fuel Cycles in GT-MHR, N. Kodochigov, Yu. Sukharev, E. Marova (OKB Mechanical Engineering), N. Ponomarev-Stepnoy, E. Glushkov, P. Fomichenko (RRC Kurchatov Institute)

Session 5C Critical and Subcritical Experiments ***Columbus H***

Session Organizers: Russell D. Mosteller (LANL), Hironobu Unesaki (Kyoto University).
Session Chairs: Russell D. Mosteller (LANL), Alain Santamarina (CEA Cadarache).

8:30 AM Criticality Analysis of Highly Enriched Uranium / Thorium Fueled Thermal Spectrum Cores of Kyoto University Critical Assembly, Hironobu Unesaki, Tsuyoshi Misawa, Chihiro Ichihara, Keiji Kobayashi, Hiroshi Nakamura, Seiji Shiroya (Kyoto University), Kazuhiko Kudo (Kyushu University)

8:50 AM Analysis of Criticality Change with Time for MOX Cores, Ken Nakajima (Kyoto University), Takenori Suzaki (Japan Atomic Energy Research Institute)

9:10 AM Subcritical Experiments in Uranium-Fueled Core with Central Test Zone of Tungsten, Tsuyoshi Yamane, Shigeaki Okajima (Japan Atomic Energy Research Institute)

9:30 AM Re-Evaluation of SEFOR Doppler Experiments and Analyses with JNC and ERANOS systems, Taira Hazama (Japan Nuclear Cycle Development Institute), Jean Tommasi (CEA)

9:50 AM The TRADE Source Multiplication Experiments, G. Imel, D. Naberejnev, G. Palmiotti (ANL), H. Philibert, G. Granget, L. Mandard, R. Soule, P. Fougeras (CEA), J. C. Steckmeyer, F. R. Lecolley (LPC-ENSICAEN-IN2P3-CNRS), M. Carta, R. Rosa, A. Grossi, S. Monti, V. Peluso, M. Sarotto (ENEA)

10:10 AM *Break*

10:30 AM Study of the Influence of a Pulsed Source in the Kinetics Measurements in a Subcritical System, Yolanda Rugama (CEA), George Imel (ANL)

10:50 AM The MUSE4 Pulsed Neutron Source Experiments, E. M. Gonzalez-Romero, D. Villamarin, M. Embid, M. C. Vicente (Centro de Estudios Energéticos Medioambientales y Tecnológicos), C. Destouches, P. Chaussonnet, J. M. Laurens, F. Mellier (CEA)

11:10 AM Reactivity Measurements and Neutron Spectrometry in the MUSE-4 Experiment, Annick Billebaud, Joachim Vollaire, Roger Brissot, Daniel Heuer, Christian Le Brun, Eric Liatard, Jean-marie Loiseaux, Olivier Meplan, Elsa Merle-Lucotte, Alexis Nuttin, Fabien Perdu (CNRS), Cristophe Destouches, Pascal Chaussonnet, Jean-Marc Laurens, Yolanda Rugama (CEA)

11:30 AM Some Experimental Results from the Last Phases of the MUSE Program, Frederic Mellier (CEA), George Imel (ANL), Yolanda Rugama (Technological University of Delft), Christian Jammes (CEA), Jean-françois Lebrat (CEA)

Session 5D Reactor Analysis Methods ***Grand D-South***

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Elmer Lewis (Northwestern University).

8:30 AM Reactor Core Simulations in Canada, Robert Roy, Jean Koclas (École Polytechnique de Montréal), Wei Shen, Dave Jenkins, Dimitar Altiparmakov, Benjamin Rouben (Atomic Energy of Canada Limited)

8:50 AM A New Method for the Treatment of Strong Local Heterogeneities and Its Application to the PHEBUS Experimental Facility, Enrico Girardi, Jean-Michel Ruggieri, Patricia Sireta, Guillaume Ritter (CEA)

9:10 AM Mixed-Hybrid Methods for the Linear Transport Equation, Serge Van Criekingen, Elmer E. Lewis (Northwestern University), R. Beauwens (Free University of Brussels)

9:30 AM Adaptative Solution of the Multigroup Diffusion Equation on Irregular Structured Grids Using a Non Conforming Finite Element Method Formulation, Jean C. Ragusa (CEA)

9:50 AM The Variational Nodal Method in R-Z Geometry, Hui Zhang (Northwestern University), Elmer E. Lewis (Northwestern University)

10:10 AM *Break*

10:30 AM Solving the Neutron Diffusion Equation on Combinatorial Geometry Computational Cells for Reactor Physics Calculations, Yousry Youssef Azmy (Pennsylvania State University)

10:50 AM Development of Hybrid Core Calculation System Using 2-D Full-Core Heterogeneous Transport Calculation and 3-D Advanced Nodal Calculation, Masaaki Mori, Naoki Sugimura, Masayuki Hijiya, Tadashi Ushio (Nuclear Engineering, Limited), Yasushi Arakawa (The Kansai Electric Power Co., Inc.)

11:10 AM A First-Order Spherical Harmonics Formulation Compatible with the Variational Nodal Method, Micheal A. Smith, Giuseppe Palmiotti, Won Sik Yang (ANL), Elmer E. Lewis (Northwestern University)

11:30 AM Investigating the Use of 3-D Deterministic Transport for Core Safety Analysis, Hans D. Gougar, D. Scott Lucas, Paul A. Roth (INEEL), Todd A. Wareing, Greg Failla, John McGhee (Radion Technologies)

Wednesday, April 28, 2004, 12:00 P.M.

Luncheon ***Columbus IJKL***

Emerging Areas in Reactor Physics, Cecil Parks (Oak Ridge National Laboratory)

Wednesday, April 28, 2004, 1:30 P.M.

Session 6A Advanced Reactor Designs ***Columbus EF***

Session Organizers: Bojan Petrovic (Westinghouse), Mario Carelli (Westinghouse), Marc Delpech (CEA).
Session Chair: Bojan Petrovic (Westinghouse).

1:30 PM Feasibility and Configuration of a Mixed Spectrum Supercritical Water Reactor, Taek K. Kim (ANL), Paul Wilson, Po Hu, Rachna Jain (University of Wisconsin-Madison)

1:50 PM Experimental Study on Reduced Moderation BWR with Advanced Recycle System (BARS), Kouji Hiraiwa, Kenichi Yoshioka, Yasushi Yamamoto, Miyuki Akiba, Mitsuaki Yamaoka (Toshiba), Junji Mimatsu (Gifu University)

2:10 PM A Core Design for a Single Fuel Enrichment in a Self-Sustaining Lead-Cooled Reactor, Yonghee Kim, Choong Ho Cho, Sang Ji Kim, Tae Yung Song (Korea Atomic Energy Research Institute)

2:30 PM Preliminary Neutronics Design Studies of a Lead Cooled, Small Modular Reactor, Micheal A. Smith, Won Sik Yang, Temitope A. Taiwo (ANL)

2:50 PM Physics and Safety Studies of a Low Conversion Ratio Sodium Cooled Fast Reactor, Micheal A. Smith, James E. Cahalan, Robert N. Hill, Floyd E. Dunn (ANL)

3:10 PM *Break*

3:30 PM Preliminary Neutronics Design Studies for a 400 MWt STAR-LM, Gerardo Aliberti, Won Sik Yang, John Stillman, Robert N. Hill (ANL)

3:50 PM PDS-XADS LBE and Gas-Cooled Concepts: Neutronic Parameters Comparison, Sandro Pelloni (Paul Scherrer Institut)

4:10 PM Study of Accelerator Transient on ADS Operation Using TRACY(TRAnsient experiment Critical facilitY), Satoshi Gunji (Tohoku University)

4:30 PM Activities of Working Party on 'Subcritical Core of Accelerator-Driven System' in the Research Committee on Reactor Physics of AESJ and JAERI, Tomohiko Iwasaki (Tohoku University), Kazufumi Tsujimoto, Kenji Nishihara (JAERI), Yasunori Kitamura (Nagoya University)

Session 6B Nuclear Safety ***Columbus G***

Session Organizers: Kostadin N. Ivanov (Pennsylvania State University), Werner Maschek (Karlsruhe).
Session Chairs: David Diamond (BNL), Siegfried Langenbuch (GRS).

1:30 PM Analysis of Boron Dilution Transients in PWRs, David Diamond (BNL), Blair Bromley (Atomic Energy of Canada, Ltd.), Arnold Aronson (BNL)

1:50 PM Uncertainty Evaluation of the Results of the MSLB Benchmark by CIAU Methodology, Alessandro Petruzzi (Pennsylvania State University), Francesco S. D'auria (University of Pisa), Kostadin Ivanov (Pennsylvania State University)

2:10 PM Improving the Computation Efficiency of COBRA-TF for LWR Safety Analysis of Large Problems, Diana Cuervo (Polytechnic University of Madrid), Maria Nikolova Avramova, Kostadin Ivanov (Pennsylvania State University)

2:30 PM POLCA-T Simulation of OECD/NRC BWR Turbine Trip Benchmark Exercise 3 Best Estimate Scenario TT2 Test and Four Extreme Scenarios, Dobromir Panayotov (Westinghouse Atom AB)

2:50 PM Localized Void Feedback Effects Under Single Rod Drop Transient in BWR, Akitoshi Hotta, Hiroshi Shirai, Shinya Mizokami (TEPCO Systems Corporation)

3:10 PM *Break*

3:30 PM Peach Bottom-2 Low-Flow Stability Test using Trac-Bf1/Valkin and Relap5-Mod.3.3/Parcs Codes, Rafael Miró, Ana María Sánchez, Gumersindo Verdu Martin, Damian Ginestar (Universitat Politecnica de Valencia), Francesca Maggini, Francesco D'auria (Università di Pisa)

3:50 PM Bruce 'B' Core Conversion, Evgeny Braverman, Ovidiu Nainer (Bruce Power)

4:10 PM Coupled Neutronics-Thermalhydraulics Calculations for the Safety Analysis of the PBMR, Bismark Mzubanzi Tyobeka, Kostadin Ivanov (Pennsylvania State University)

Session 6C Physics Code Validation ***Columbus H***

Session Organizers: Alain Santamarina (CEA Cadarache), Robert Jacqmin (CEA Cadarache), Richard Sanchez (CEA).
Session Chair: Alain Santamarina (CEA Cadarache), Farzad Rahnema (Georgia Institute of Technology).

1:30 PM Experimental Validation of APOLLO2 Code for High Burnup MOX Fuels. JEF2.2 Results and JEFF3.0 Improvements, David Bernard, Alain Santamarina (CEA Cadarache), Anne-Marie Malvagi (Electricite de France), Lucien Daudin (FRAMATOME)

1:50 PM Assessment of CASMO-4 Predictions of the Isotopic Inventory of High Burn-Up MOX Fuel, Rafael Macian, Martin A. Zimmermann, Rakesh Chawla (Paul Scherrer Institut)

2:10 PM PANDA Code Application to the OECD/NEA 2D/3D-MOX Assembly Benchmark Calculations, Philippe P. Humbert (CEA)

2:30 PM Benchmark Analysis of the DeCART MOC Code with the VENUS-2 Critical Experiment, Zhaopeng Zhong, Thomas J. Downar (Purdue University), Han Gyu Joo, Jin Young Cho (KAERI)

2:50 PM Monte Carlo Analysis of High Moderation 100% MOX BWR Cores using JEF2 and JENDL3 Nuclear Data, Olivier Litaize, Alain Santamarina, Morgan Hervault, Stephane Cathalau, Philippe Fougeras, Patrick Blaise (CEA), Toru Yamamoto, Ryoji Kanda, Masaru Sasagawa, Tsukasa Kikuchi (NUPEC)

3:10 PM *Break*

3:30 PM Experimental Validation of Pin Power Distributions for a BWR Assembly with Hafnium Control Blades, Fabian Jatuff, P. Grimm, M. Murphy, R. Seiler (Paul Scherrer Institut), R. Jacot-Guillarmod, J. Krouthen, T. Williams (Westinghouse Electric Sweden AB), R. Chawla (PSI/Swiss Federal Institute of Technology)

3:50 PM Validation of Integrated Burnup Code System SWAT2 by the Analysis of Isotopic Composition Data of Spent Nuclear Fuel, Kenya Suyama (JAERI), Hiroki Mochizuki (The Japan Research Institute Limited), Hiroshi Okuno, Yoshinori Miyoshi (JAERI)

4:10 PM Verification of Lattice Analysis Method through BWR UO2 PIE Data Analysis, Toru Yamamoto (Japan Nuclear Energy Safety Organization), M. Sasagawa (NUPEC), K. Kawashima (Japan Nuclear Energy Safety Organization

4:30 PM Uncertainty Analysis Applied to Fuel Depletion Calculations, Rafael Macian, Martin A. Zimmermann, Rakesh Chawla (Paul Scherrer Institut)

Session 6D Reactor Analysis Methods ***Grand D-North***

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Won Sik Yang (ANL), Thomas J. Downar (Purdue University).

1:30 PM Neutronics Codes Currently Used in Japan for Fast and Thermal Reactor Applications, Toshikazu Takeda (Osaka University)

1:50 PM HORUS3D Code Package Development and Validation for the JHR Modeling, G. P. Willermoz, A. Aggery, D. Blanchet, S. Cathalau, C. Chichoux, J. Di-Salvo, C. Doderlein, D. Gallo, F. Gaudier, N. Huot, S. Loubière, B. Noël, H. Serviere (CEA)

2:10 PM On Some Features of Quasi-Static Schemes in Reactor Kinetics, Piero Ravetto, Sandra Dulla (Politecnico Di Torino), Matteo M. Rostagno (ENEA-Bologna)

2:30 PM Albedo Conditions for Multigroup Anisotropic Scattering Models of Nuclear Reactors, Marcos P. De Abreu (State University of Rio de Janeiro)

2:50 PM An Algebraic Multgrid Resolution Strategy for the DPn Synthetic Acceleration Method, S. Santandrea (CEA)

3:10 PM *Break*

3:30 PM Convergence Analysis of 2-D/1-D Coupling Strategies for Diffusion Equation, Hyun Chul Lee, Deokjung Lee, Thomas J. Downar (Purdue University)

3:50 PM Numerical Convergence Analysis of the Nonlinear CMFD Method for Three-Dimensional Two-Group LWR Diffusion Problems, Deokjung Lee, Thomas J. Downar (Purdue University), Yonghee Kim (KAERI)

4:10 PM An Approach of Super-Element Sweeping for the Solution of Neutron Transport Equation in Heterogeneous Geometry, Nam Cho, Gil Soo Lee (KAIST)

4:30 PM Efficient Hybrid NEM/BEM Transient Method, Dianna M. Hahn, Kostadin Ivanov (Pennsylvania State University), Piero Ravetto (Politecnico Di Torino), Matteo Rostagno (ENEA-Bologna)

Thursday, April 29, 2004, 8:30 A.M.

Plenary Session - Efforts in Code Development and Reactor Modeling ***Columbus IJKL***
Session Chair: Enrique M. Gonzalez-Romero (CEEMT)

Neutronics Codes Currently Used in Japan for Fast and Thermal Reactor Applications, Toshikazu Takeda (Osaka University)

Monte-Carlo Methods and MCNP Code Developments, Forrest Brown (Los Alamos National Laboratory)

Thursday, April 29, 2004, 10:00 A.M.

Session 7A Non-Conventional Reactors ***Columbus EF***

Session Organizer: Jan Leen Kloosterman (Delft University of Technology).
Session Chair: Danny Lathouwers (Delft University of Technology).

10:00 AM Simulation of Caliban Reactor Burst Wait Time and Initiation Probability using a Point Reactor Model and PANDA Code, Philippe P. Humbert, Boukhmes Mechitoua (CEA)

10:20 AM Passive Decay Heat Removal in a Fluidized Bed Nuclear Reactor, A. Agung, D. Lathouwers, T. Van Der Hagen, H. Van Dam (Delft University of Technology), C. C. Pain (Imperial College), C. R. E. de Oliveira (Georgia Institute of Technology), A. J. Goddard, M. D. Eaton, J. L. Gomes, B. Miles (Imperial College)

10:40 AM Molten Salt Reactors and Possible Scenarios for Future Nuclear Power Deployment, Elsa Merle-Lucotte, Daniel Heuer, Ludovic Mathieu, Jean-Marie Loiseaux, Annick Billebaud, Roger Brissot, Christian Le Brun, Olivier Laulan, Eric Liatard, Olivier Meplan, Alexis Nuttin, Sylvain David, Franco Michel-Sendis (Centre National de la Recherche Scientifique)

11:00 AM Closing the PWR Fuel Cycle with a Molten-Salt Incinerator, Radim Vocka (Nuclear Research Institute Rez plc.)

11:20 AM Monte Carlo Calculation of the Effects of Delayed Neutron Precursor Transport in Molten Salt Reactors, J. Kophazi, Mate Szieberth, Sando Feher, Szabolcs Czifrus (Budapest University of Technology and Economics), P.F.A. de Leege (Delft University of Technology)

11:40 AM Studies of Physical Features of Cascade Subcritical Molten Salt Reactor with External Neutron Source, Vladimir A. Nevinitsa, Pavel N. Alekseev, Peter A. Fomichenko, Alexander V. Vasiliev, Anatoly A. Dudnikov, Aleksey A. Sedov, Stanislav A. Subbotin (Russian Research Center Kurchatov Institute), Andrey M. Voloshchenko (Keldysh Institute of Applied Mathematics)

Session 7B Nuclear Data ***Columbus CD***

Session Organizers: Luiz Leal (ORNL), Mark Chadwick (LANL).
Session Chairs: Russell Mosteller (LANL), Alain Santamarina (CEA Cadarache).

10:00 AM New Neutron Cross Section Measurements at ORELA for Improved Nuclear Data, Klaus H. Guber, Luiz C. Leal, Royce O. Sayer, Paul E. Koehler, T. E. Valentine, H. Derrien, J. A. Harvey (ORNL)

10:20 AM Measurement and Calculation of the 233Pa Fission Cross-Section for Advanced Fuel Cycles, Franz-Josef Hambsch, Stephan Oberstedt, Volker Fritsch (EC-JRC-IRMM), Fredrik Tovesson, Andreas Oberstedt (Oerebro University), Birger Fogelberg, Elisabet Ramstroem (Uppsala University), Georghita Vladuca, Anabella Tudora, D. Filipescu (Bucharest University)

10:40 AM Scattering Law Data for Graphite in Gas Cooled High Temperature Reactors, Wolfgang Bernnat, Margarete Mattes, Juergen Keinert (University of Stuttgart)

11:00 AM An Unresolved Resonance Evaluation for U-235, L. Leal, H. Derrien, N. M. Larson (Oak Ridge National Laboratory)

11:20 AM Status of a New Evaluation of the Neutron Resonance Parameters of 238U at ORNL, Herve Derrien, L.C. Leal, N.M. Larson (ORNL)

11:40 AM JEF2.2 Nuclear Data Statistical Adjustment using Post-Irradiation Experiments, A. Courcelle, A. Santamarina, F. Bocquet, G. Combes, C. Mounier, G. Willermoz (CEA)

Session 7C Nuclear Safety ***Columbus KL***

Session Organizers: Kostadin N. Ivanov (Pennsylvania State University), Werner Maschek (Karlsruhe).
Session Chairs: Dobromir Panayotov (Westinghouse Atom AB), Tim Newton (Serco Assurance).

10:00 AM	OECD/DOE/CEA VVER-1000 Coolant Transient (V1000CT) Benchmark for Assessing Coupled Neutronics/Thermal-Hydraulics System Codes for VVER-1000 RIA Analysis, Boyan D. Ivanov, Kostadin N. Ivanov (Pennsylvania State University), Eric Denis Royer, Sylvie Aniel (CEA), Nikola Kolev, Pavlin Groudev (INRNE)
10:20 AM	Uncertainty and Sensitivity Analysis Applied to Coupled Code Calculations for a VVER Plant Transient, S. Langenbuch, B. Krzykacz-Hausmann, K.-D. Schmidt (Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH)
10:40 AM	Study of Nuclear Fuel Behavior with Coupled 3D Neutronics/Thermal-Hydraulic Codes, Carlo Parisi (University of Pisa), Kostadin Ivanov (Pennsylvania State University), Francesco S. D'auria (University of Pisa)
11:00 AM	TRAC-M/AAA Code Assessment for Transient Analysis of Pb-Bi Cooled Fast-Spectrum Reactor Systems, Konstantin Mikityuk, Paul Coddington, Rakesh Chawla (Paul Scherrer Institut)
11:20 AM	Source and Reactivity Perturbations in Accelerator Driven Systems with Conventional MOX and Advanced Fertile Free Fuel, Xue-nong Chen, Tohru Suzuki, Andrei Rineiski, Claudia Matzerath Boccaccini, Werner Maschek (Forschungszentrum Karlsruhe), Peter Smith (Serco Assurance)
11:40 AM	Safety Characteristics of Candidate Oxide Fuels for Accelerator Driven Transmuters, Tim Newton, Peter Smith (Serco Assurance)

Session 7D Physics and Modeling of Research Reactors in INIE's Big-10 Consortium ***Columbus IJ***

Session Organizers: Yousry Azmy (Pennsylvania State University), Rizwan Uddin (University of Illinois at Urbana-Champaign). Session Chairs: Yousry Azmy (Pennsylvania State University), Rizwan Uddin (University of Illinois at Urbana-Champaign).

10:00 AM	Beam Calculation for TRIGA Reactor, Federico E. Teruel, Rizwan Uddin (University of Illinois at Urbana-Champaign)
10:20 AM	Thermal Neutron Time-of-Flight Spectroscopy at Penn State using a Single-Disk Chopper, John Niederhaus (University of Wisconsin-Madison), Jack Brenizer, Kenan Unlu (Pennsylvania State University)
10:40 AM	UIUC’s Contribution to Big-10’s INIE Project, Yuxiang Gu, Nick Karancevic, Koji Sugawara, Yizhou Yan, James Stubbins, Rizwan Uddin (University of Illinois at Urbana-Champaign)
11:00 AM	Modeling of Existing Beam-Port Facility at PSU Breazeale Reactor by Using MCNP, Baris Sarikaya, Fatih Alim, Kostadin N. Ivanov, Kenan Unlu, Jack Brenizer, Yousry Azmy (Pennsylvania State University)
11:20 AM	AGENT Code: Open-Architecture Analysis and Configuration of Research Reactors – Neutron Transport Modeling with Numerical Examples, Tatjana Jevremovic, Hyun Chul Lee, Kevin Retzke, Yefei Peng, Mathieu Hursin (Purdue University)
11:40 AM	A Three Dimensional Two Energy Group Coupled Reactor Physics and Thermal Hydraulics Code (M32) - A Tool for Student Design Studies, James Brushwood, J. P. Alcock, A. Thompson, Philip A. Beeley (University of Surrey), J. Moorby (Private Consultant)

Thursday, April 29, 2004, 1:00 P.M.

Session 8A Fuel Cycle Physics ***Columbus EF***

Session Organizers: Alain Zaetta (CEA Cadarache), Michael Todosow (BNL).
Session Chairs: Michael Todosow (BNL), John Stillman (ANL).

1:00 PM	A Study on High-Intensity Radiation Protection of MOX-Fuel Doped with Protactinium, E. Kryuchkov, V. B. Glebov, V. A. Apse, A. N. Shmelev (Moscow Engineering Physics Institute (State University)), M. Saito, V. V. Artisyuk (Tokyo Institute of Technology)
1:20 PM	Optimization Studies for Seed-and-Blanket Unit (SBU) Fuel Assemblies in PWRs, Blair P. Bromley, Michael Todosow, Arnold Aronson (BNL), Alex Galperin (Ben-Gurion University of the Negev)
1:40 PM	Reactivity and Neutron Emission Measurements of Burnt PWR Fuel Rod Samples in LWR-PROTEUS Phase II, M. F. Murphy, F. Jatuff, P. Grimm, R. Seiler, R. Brogli (Paul Scherrer Institut), G. Meier (Kernkraftwerk Gosgen-Daniken AG), H.-D. Berger (Framatome ANP), R. Chawla (PSI/Swiss Federal Institute of Technology)
2:00 PM	Design and Analysis of Molten Salt Reactor Fueled by TRU from LWR, Massimiliano Fratoni, David Barnes, Ehud Greenspan (University of California, Berkeley), Augusto Gandini (University of Rome)

2:20 PM On the Capability of SMORES to Account for Self-Shielding in Search for Maximum k_{eff}, Yonathan Karni, Ehud Greenspan (University of California), Sedat Goluoglu, Calvin Hopper (ORNL)

2:20 PM Concept for a Thermal-Hydraulic 3D Parallel Channel Core Model, Alois Hoeld (Private (Retired))

2:40 PM Reactivity Effects due to the Beryllium Poisoning of BR2 - MCNP Calculations, Silva Kalcheva, Bernard Ponsard, Edgar Koonen (SCK-CEN (Belgian Nuclear Research Center))

Session 8B Nuclear Data ***Columbus CD***

Session Organizers: Luiz Leal (ORNL), Mark Chadwick (LANL).
Session Chairs: Michael Dunn (ORNL), Soo-Youl Oh (KAERI).

1:00 PM An Assessment of ENDF/B-VI Releases Using the MCNP Criticality Validation Suite, Russell D. Mosteller (LANL)

1:20 PM Generation and Performance of a Multigroup Coupled Neutron-Gamma Cross-Section Library for Deterministic and Monte Carlo Borehole Logging Analysis, J. Eduard Hoogenboom (Delft University of Technology), Ivo Kodeli (OECD), Daniel L. Aldama (Jozef Stefan Institute), Piet F.A. de Leege, David Legrady (Delft University of Technology)

1:40 PM On the Importance of a New Formula for the Double Differential Scattering Kernel, Ron Dagan (FZK-Forschungzentrum Karlsruhe)

2:00 PM Effect of Energy Self-Shielding on Reactor Benchmark Problems, F. Arzu Alpan, Luiz Leal (ORNL), Arnaud Courcelle (CEA)

2:20 PM Development of a Methodology for Analysis of the Impact of Modifying Neutron Cross Sections, Michael T. Wenner, Alireza Haghighat (University of Florida), James M. Adams, Allan D. Carlson (National Institute of Standards and Technology), Steven M. Grimes, Thomas N. Massey (Ohio University)

2:40 PM The Fission Spectrum Uncertainty, Bryan Broadhead (University of Tennessee, Battelle), Jekutiel Jehudah Wagschal (Hebrew University of Jerusalem)

3:00 PM The TRADE Experiment: Importance of Neutron Cross-Sections for Transmutation, Yacine Kadi, Adonai Herrera-Martinez, Marcus Dahlfors (European Organization for Nuclear Research)

3:20 PM Box-Cox Transformation for Resolving the Peelle's Pertinent Puzzle in a Curve Fitting, Soo-Youl Oh, Chul-Gyo Seo (Korea Atomic Energy Research Institute)

Session 8C Physics Code Validation ***Columbus KL***

Session Organizers: Alain Santamarina (CEA Cadarache), Robert Jacqmin (CEA Cadarache), Richard Sanchez (CEA).
Session Chair: Robert Jacqmin (CEA Cadarache).

1:00 PM Benchmark Calculation of WIMS/RFSP against Wolsong Nuclear Power Plants 3 and 4 Physics Measurement Data, Hangbok Choi, Donghwan Park (Korea Atomic Energy Research Institute)

1:20 PM Qualification of MCNP Coolant Void Reactivity Calculations using ZED-2 Measurements, Ken Kozier (Atomic Energy of Canada Ltd.)

1:40 PM DYN3D Calculations for the V-1000 Test Facility and Comparisons with the Measurements, Ulrich Grundmann (Forschungszentrum Rossendorf (FZR)), Siegfried Mittag (Forschungszentrum Rossendorf Inc.)

2:00 PM Validation of SCALE4.4a/CSAS25 for Nuclear Criticality Safety Analyses, Robert H. Smith (BWXT Y-12), John Declue, Cris Worley (Y-12 National Security Complex)

2:20 PM The Nuclear Heating Calculation Scheme for Material Testing in the Future Jules Horowitz Reactor, Nicolas Huot, Arnaud Courcelle, David Blanchet, Jacques Di-Salvo, Cristophe Doderlein, Hugues Serviere, Guy Pierre Willermoz (CEA - Cadarache), Alain Aggery (CEA - Saclay)

2:40 PM Analysis of the ZPR-9 Gas-Cooled Fast Reactor Experiments Using JEF-2.2 Data and the ERANOS Code System, Jean Tommasi (CEA)

3:00 PM Spatially and Temperature Dependent Dancoff Method for LWR Lattice Physics Code, Hideki Matsumoto (Mitsubishi Heavy Industries, Ltd. (Japan)), Mohamed Ouisloumen (Westinghouse Electric Company), Yoshihisa Tahara (Mitsubishi Heavy Industries, Ltd. (Japan))

Session 8D Reactor Analysis Methods ***Columbus IJ***

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Cecil Parks (ORNL).

1:00 PM French Calculation Schemes for Light Water Reactor Analysis, Alain Santamarina (CEA), Claire Collignon (EDF/R&D), Christian Garat (FRAMATOME-ANP SAS)

1:20 PM Irradiation Experiment Analysis for Cross Section Validation, Luigi Mercatali, Giuseppe Palmiotti, Massimo Salvatores (ANL), Jean Tommasi (CEA)

1:40 PM	Methodologies for Treatment of Spectral Effects at Core-Reflector Interfaces in Fast Neutron Systems, Gerardo Aliberti, Giuseppe Palmiotti, Massimo Salvatores (ANL), J. F. Lebrat, Jean Tommasi, R. Jacqmin (CEA)
2:00 PM	Effect of Pellet Radial Power and Temperature Distribution on Fuel Assembly Neutronics, Mohamed Ouisloumen, (Westinghouse Electric Corporation), Emily Wolters (Westinghouse Company), Hideki Matsumoto (Mitsubishi Heavy Industries, Ltd)
2:20 PM	A Comparison of Binary Stochastic Media Transport Models in 'Solid-Void' Mixtures, Ian Davis, Todd S. Palmer (Oregon State University), Edward W. Larsen (University of Michigan)
2:40 PM	Development of an Object Oriented Nodal Code using the Refined AFEN Derived from the Method of Component Decomposition, Jae Man Noh, Jae-Woon Yoo, Hyung-Kook Joo (Korea Atomic Energy Research Institute)
3:00 PM	Applications of Modal-Local Analysis for Source-Driven Subcritical Systems, Viktoriya V. Kulik, John Lee (University of Michigan)
3:20 PM	Development of a Polynomial Nodal Model for the Multi Group Diffusion Equation in 2-D, A. Shojaei, J. Khorsandi (Esfahan Nuclear Technology Center, Iran)

Technical Program Abstracts

Abstacts for each paper in the technical program are provided on the following pages. For your convenience, the abstracts are organized by session as follows:

The Numerical Nuclear Reactor for High Fidelity Integrated Simulation of Neutronic, Thermal-Hydraulic and Thermo Mechanical Phenomena – Project Overview

D. P. Weber[1], T. Sofu[1], P. A. Pfeiffer[1], W. S. Yang[1], T. A. Taiwo[1], H. G. Joo[2], J. Y. Cho[2], K. S. Kim[2], T. H. Chun[2], T. J. Downar[3], J. W. Thomas[3], Z. Zhong[3], C. H. Kim[4], B. S. Han[4]
[1]*Argonne National Laboratory (208), 9700 S. Cass Ave., Argonne, IL 60439 USA*
[2]*Korea Atomic Energy Research Institute, Yuseong, Daejeon 305-600, Korea*
[3]*Purdue University, 1290 Nuclear Engineering, Building, West Lafayette, IN 47907-1290*
[4]*Seoul National University, San 56-1, Shillim-dong, Gwanak-gu, Seoul, Korea*

As part of a US-ROK collaborative I-NERI project, a comprehensive high fidelity reactor core modeling capability is being developed for detailed analysis of current and advanced reactor designs. The work involves the coupling of advanced numerical models such as computational fluid dynamics (CFD) for thermal hydraulic calculations, whole core discrete integral transport for neutronics calculations, and thermo-mechanical techniques for structural calculations. The product code has been designed to run on parallel high performance computers. This integrated simulation capability will provide a verifiable computational tool to perform intensive studies on the operational and safety characteristics of various design alternatives and to compare the results obtained with presently available tools to those from this high fidelity capability. This paper provides an overview of the project and a summary of the key elements of the integrated code.

Methods and Performance of a Three-Dimensional Whole-Core Transport Code DeCART

Han Gyu Joo[1], Jin Young Cho[1], Kang Seog Kim[1], Chung Chan Lee[1] and Sung Quun Zee[1]
[1]*Korea Atomic Energy Research Institute, Yuseong, Daejeon, 305-600, Korea*

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DeCART is a three-dimensional whole-core transport code capable of performing direct core calculations at power generating conditions without involving *a priori* homogenized few-group constant generation. In this paper, the methods of DeCART, which are characterized by the planar method of characteristics (MOC) solutions, the cell based coarse mesh finite difference (CMFD) formulation, the subgroup method for resonance treatment and subpin level thermal feedback, are presented as a whole. The performance of the code in the aspect of solution accuracy and computing speed is then examined using the applications to the C5G7MOX benchmark and its modified rodded variation problems and also to a three-dimensional core case involving thermal feedback. The examination indicates that accurate direct whole core calculations with subpin level thermal feedback for practical PWR problems are quite possible on affordable LINUX clusters within a time span of a few hours.

Consistent Comparison of Monte Carlo and Whole-Core Transport Solutions for Cores with Thermal Feedback

Han Gyu Joo[1], Jin Young Cho[1], Kyo Youn Kim[1], Moon Hee Chang[1], Beom Seok Han[2] and Chang Hyo Kim[2]
[1]*Korea Atomic Energy Research Institute, Yuseong, Daejeon 305-600, Korea*
[2]*Seoul National University, San 56-1, Shillim-dong, Seoul, 159-741, Korea*

For systematic and consistent comparison of Monte Carlo and whole-core transport solutions in various core states including power generating conditions, a test problem set that spans from two-dimensional uniform temperature pin cell problems to three-dimensional core problems involving thermal feedback is solved by a continuous energy Monte Carlo code MCCARD and a multigroup whole-core transport code DeCART. The neutron spectra, k-effective, pin-wise power distribution, fuel temperature distribution, and Doppler coefficients obtained from the two solutions are compared taking the MCCARD solution as the reference. For the uniform temperature problems, excellent agreement between the two solutions is observed in every solution aspect. The pin power distribution error is less than 1% and the k-effective error is within 100 pcm in most cases. For the problems with thermal feedback, the discrepancy becomes

larger, yet within the tolerable range. In the hot-full-power minicore calculation, a maximum of 3.7% error in the radial pin power distribution and the k-effective error of about 260 pcm are observed. Through this comparison, it is demonstrated that accurate multigroup direct whole-core calculations are possible even at power generating conditions with a much less computing time than the corresponding Monte Carlo calculations.

Evaluation of Turbulence Models for Flow and Heat Transfer in Fuel Rod Bundle Geometries

T. Sofu[1], T. H. Chun[2] and Dr. W.K. In[2]
[1]*Argonne National Laboratory, 9700 S. Cass Ave., Argonne, IL 60439 USA*
[2]*Korea Atomic Energy Research Institute, Yusung, Daejeon 305-600, Korea*

One of the objectives of the US-ROK collaborative I-NERI project known as the “Numerical Reactor” is an assessment of commercial Computational Fluid Dynamics (CFD) analysis capabilities for high-fidelity thermal-hydraulic analysis of current and advanced reactor designs. More specifically, the work involves evaluation of common turbulence models in terms of their ability to calculate the flow and heat transfer for simple fuel rod bundle configurations. The evaluations have so far focused mostly on Reynolds-Averaged Navier-Stokes (RANS) models--including the standard k-ε model, non-linear (quadratic and cubic) k-ε models, and the renormalization-group (RNG) variant. The second-order moment closure models such as the differential Reynolds stress model (RSM) have also been considered.

For a set of different rod bundle configurations with and without mixing promoters, the code predictions have been compared with data from literature on experiments with and without heat transfer. In general, the accurate predictions for distribution of turbulent structures in the subchannel have not been demonstrated with the most common standard k-ε model. However, the non-linear RANS models, RSM, and the double-layer approach based on the use of low-Re number k-ε model in the boundary layer seem to improve the predictions of turbulence intensity and fluid temperature distributions noticeably. The comparisons also indicate a need for assessments of unsteady turbulence models to simulate additional turbulent motion of large-scale eddy fluctuations in the coolant channels.

Methodology for Coupling Computational Fluid Dynamics and Integral Transport Neutronics

J.W. Thomas[1], Z. Zhong[1], T. Sofu[2], T.J. Downar[1]
[1]*Purdue University, West Lafayette, IN 47906 USA*
[2]*Argonne National Laboratory, 9700 S. Cass Ave., Argonne, IL 60439 USA*

The CFD code STAR-CD was coupled to the integral transport code DeCART in order to provide high-fidelity, full physics reactor simulations. An interface program was developed to perform the tasks of mapping the STAR-CD mesh to the DeCART mesh, managing all communication between STAR-CD and DeCART, and monitoring the convergence of the coupled calculations. The interface software was validated by comparing coupled calculation results with those obtained using an independently developed interface program. An investigation into the convergence characteristics of coupled calculations was performed using several test models on a multiprocessor LINUX cluster. The results indicate that the optimal convergence of the coupled field calculation depends on several factors, to include the tolerance of the STAR-CD solution and the number of DeCART transport sweeps performed before exchanging data between codes. Results for a 3D, multi-assembly PWR problem on 12 PEs of the LINUX cluster indicate the best performance is achieved when the STAR-CD tolerance and number of DeCART transport sweeps are chosen such that the two fields converge at approximately the same rate.

Coupled Calculations Using the Numerical Nuclear Reactor for Integrated Simulation of Neutronic and Thermal-Hydraulic Phenomena

D. P. Weber[1], T. Sofu[1], W. S. Yang[1], T. J. Downar[2], J. W. Thomas[2], Z. Zhong[2], H. G. Joo[3]
[1]Argonne National Laboratory (208), 9700 S. Cass Ave., Argonne, IL 60439 USA
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[3]Korea Atomic Energy Research Institute, Yuseong, Daejeon 305-600, Korea

The Numerical Nuclear Reactor is a collaborative US-ROK I-NERI project to develop a comprehensive high fidelity reactor core modeling capability for detailed analysis of current and advanced reactor design. High fidelity models of thermal-hydraulics and neutronics have been incorporated into the initial version of the integrated code. Computational fluid dynamics (CFD) methods, based on the STAR-CD commercial CFD code, are used for the thermal-hydraulic analysis, while neutronics calculations are being performed with the whole core discrete integral transport code, DeCART. A robust coupling strategy for integrated, high-fidelity thermal-hydraulic/neutronic calculations has been developed. Examples of the coupled code analysis capability are provided for single pin, multi-pin and mini-core configurations, as well as estimates of computational requirements for integrated whole core calculations.

OSMOSE: An Experimental Program for the Qualification of Integral Cross Sections of Actinides

J.-P. Hudelot[1], R. Klann[2], P. Fougeras[1], F. Jorion[3], N. Drin[3], L. Donnet[3]
[1]CEA Cadarache, DEN/DER, 13108 Saint Paul lez Durance, France
[2]Argonne National Laboratory, 9700 South Cass Avenue, Argonne, IL 60439, USA
[3]CEA VALRHÔ, DEN/DRCP, BP171, 30207 Bagnols-sur-Cèze Cedex, France

The design of nuclear systems has shifted over the years from a “test and build” approach to a much more analytical methodology based on the many advances in computational techniques and nuclear data. To a large extent current reactors can be calculated almost as well as they can be measured. This is due in particular to the high quality nuclear data available for the few major isotopes which dominate the neutronics of these systems. Nevertheless, most future nuclear systems concepts and advanced fuels development programs use significant quantities of minor actinides to address modern day issues such as proliferation resistance and low cost. For example, high burnup fuels contain large quantities of americium and curium. Systems designed for plutonium and minor actinide burning are very sensitive to uncertainties in americium and curium data. There are also several other programs where the minor actinide data are essential. These include the Accelerator Transmutation of Waste concepts and Burnup Credit programs.

The need for better nuclear data on minor actinides have been stressed by various organizations throughout the world, and results of studies have been published which demonstrate that current data are inadequate for designing the projects under consideration [1, 2]. The first step in obtaining better nuclear data consists of measuring accurate integral data and comparing it to integrated energy dependent data: this comparison provides a direct assessment of the effect of deficiencies in the differential data.

An ambitious program between the Commissariat a l’Energie Atomique (CEA) and the U.S. Department of Energy (DOE) has been launched with the aim of measuring the integral absorption rate parameters in the MINERVE experimental facility located at the CEA Cadarache Research Center. The OSMOSE Program (Oscillation in Minerve of isotopes in “Eupraxic” Spectra) includes a complete analytical program associated with the experimental measurement program and aims at understanding and resolving potential discrepancies between calculated and measured values.

The OSMOSE program began in 2001 and will continue until 2013. The reactivity worth of samples containing separated actinides from 232Th to 245Cm will be measured by an oscillation technique with an expected accuracy better than 3%. The measurements will cover a wide range of neutron spectra, from over-moderated thermal spectra to fast spectra.

MINERVE Reactor Characterization in Support of the OSMOSE Program: Spectral Indices

R. Klann[1], J.-P. Hudelot[2], M. Antony[2], B. Micklich[1], G. Perret[1], N. Thiollay[2], G. Imel[1], J.-M. Girard[2], V. Laval[2]
[1]*Argonne National Laboratory, 9700 South Cass Avenue, Argonne, IL 60439, USA*
[2]*CEA Cadarache, DEN/DER, 13108 Saint Paul lez Durance, France*

An ambitious program between the Commissariat à l'Energie Atomique (CEA) and the U.S. Department of Energy (DOE) has been launched with the aim of measuring the integral absorption rate parameters in the MINERVE experimental facility located at the CEA Cadarache Research Center. The OSMOSE Program (Oscillation in Minerve of isotopes in "Eupraxic" Spectra) includes a complete analytical program associated with the experimental measurement program and aims at understanding and resolving potential discrepancies between calculated and measured values.

The objective of the OSMOSE program is to measure very accurate integral reaction rates in representative spectra for the actinides important to future nuclear system designs and to provide the experimental data for improving the basic nuclear data files. To gain the most information and insight from integral measurements, the essential reactor parameters should be as well characterized as possible. A description of the neutron spectra, or spectral indices, can be obtained via the use of fission chambers of different actinide isotopes.

As part of the OSMOSE program, extensive measurements have been performed to characterize the MINERVE reactor. The description and results of the spectral indices measurements of the R1-UO2 and R1-MOX reactor configurations are contained herein. Good agreement was obtained between the experimental measurements and calculations performed using the MCNP Monte Carlo code.

HTR-N Plutonium Cell Burnup Benchmark: Definition, Results & Intercomparison

J.C. Kuijper[1], N. Cerullo[5], F. Damian[2], J.L. Kloosterman[4], G. Lomonaco[5], J. Oppe[1], X. Raepsaet[2], H.J. Ruetten[3]
[1]*NRG, Petten, the Netherlands*
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[3]*FZJ, Juelich, Germany*
[4]*IRI, Delft University of Technology, Delft, the Netherlands*
[5]*Universita di Pisa, Pisa, Italy*

In order to obtain a further validation of several reactor physics code systems to be used for the analysis of High Temperature gas-cooled Reactors (HTR) for plutonium burning applications, a HTR plutonium cell burnup calculational benchmark exercise was defined. For simplicity, the benchmark configuration only consists of a single spherical, 6 cm diameter HTR ("pebble") fuel element containing coated (PuO_2) fuel particles.

The requested calculations concerned the tracking of the multiplication factor, nuclide densities in the fuel particles and other relevant parameters during the irradiation of the fuel element at constant power up to the unusually high burnup of 800 MWd/kgHM. Generally a good agreement is found between the results of 3 out of 4 participants, representing 4 out of 5 code systems. The remaining differences in results between the 3 participants can be largely attributed to differences in the modeling of the reaction paths, which are amplified by the unusually high flux levels typical to this particular benchmark.

Monte Carlo Parameter Studies and Uncertainty Analyses with MCNP5

Forrest B. Brown[1], Jeremy E. Sweezy[1], & Robert Hayes[2]
[1]*Diagnostics Applications Group (X-5), Los Alamos National Laboratory,*
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[2]*Radiological Technology, MS 452-09, WIPP Site,PO Box 2078, Carlsbad NM 88220*

A software tool called *mcnp_pstudy* has been developed to automate the setup, execution, and collection of results from a series of MCNP5 Monte Carlo calculations. This tool provides a convenient means of performing parameter studies, total uncertainty analyses, parallel job execution on clusters, stochastic geometry modeling, and other types of calculations where a series of MCNP5 jobs must be performed with varying problem input specifications.

Theoretical and Practical Study of the Variance and Efficiency of a Monte Carlo Calculation due to Russian Roulette

J. Eduard Hoogenboom
Delft University of Technology, Interfaculty Reactor Institute, Mekelweg 15, 2629 JB Delft, The Netherlands

Although Russian roulette is applied very often in Monte Carlo calculations, not much literature exists on its quantitative influence on the variance and efficiency of a Monte Carlo calculation. Elaborating on the work of Lux and Koblinger using moment equations, new relevant equations are derived to calculate the variance of a Monte Carlo simulation using Russian roulette. To demonstrate its practical application the theory is applied to a simplified transport model resulting in explicit analytical expressions for the variance of a Monte Carlo calculation and for the expected number of collisions per history. From these expressions numerical results are shown and compared with actual Monte Carlo calculations, showing an excellent agreement. By considering the number of collisions in a Monte Carlo calculation as a measure of the CPU time, also the efficiency of the Russian roulette can be studied. It opens the way for further investigations, including optimization of Russian roulette parameters.

Variance Reduction Techniques for the Monte Carlo Simulation of Neutron Noise Measurements

Máté Szieberth[1] and Jan Leen Kloosterman[2]
[1] *Budapest University of Technology and Economics, 1111 Budapest, Műegyetem rkp. 9., Hungary*
[2] *Delft University of Technology,2629 JB Delft, Mekelweg 15, The Netherlands*

This paper presents the theory and the methods to apply variance reduction techniques in the Monte-Carlo simulation of neutron noise experiments. Conventional Monte-Carlo variance reduction techniques are not applicable, because the behavior of the neutron noise depends on the collective effects of particles, and furthermore is influenced by the higher-moments of the tally distribution, which are not preserved by these methods.

The application of variance reduction methods results in weighted counts and undesired correlations from the splitted particles. A theory is developed and a correction is derived for the bias caused by the introduction of particle weights. The history splitting method is proposed to destroy the undesired correlations caused by variance reduction particle splitting. The new method makes possible the application of a variety of variance reduction techniques, but the Russian roulette game must be replaced by alternative history control methods.

The described techniques were implemented in MCNP4C and the results of preliminary calculations and comparison with analogue calculations are shown to prove the feasibility of the proposed method.

1B. Monte-Carlo Methods and Developments

Session Organizers: Forrest Brown (LANL), Richard Sanchez (CEA), Laurie Waters (LANL). Session Chairs: J. Eduard Hoogenboom (Delft University of Technology), William R. Martin (University of Michigan).

Two Dimensional Functional Expansion Tallies for Monte Carlo Simulations

David P. Griesheimer and William R. Martin

Department of Nuclear Engineering & Radiological Sciences, University of Michigan, Ann Arbor, MI 48109

Conventional Monte Carlo tallies are well suited for estimating integral quantities over large volumes. In cases where more detail is desired, such as estimating the spatial distribution of scalar flux, Monte Carlo faces several challenges. The traditional approach to obtaining these "detailed" solutions has been to divide the phase space into bins, thus creating a histogram approximation to the true solution. If many bins are used this approach can lead to large uncertainties and increased run time. One solution to this problem is the functional expansion tally (FET), a technique that allows Monte Carlo to tally not only the zeroth spatial moment of flux in each cell but also higher moments with respect to some set of basis functions. These higher moments represent additional information extracted from the random walk simulation and do not require any changes to the Monte Carlo simulation except for tallies. In this paper we present a new formulation for the 2D track length FET and its implementation in a modified version of MCNP4c. To verify our methodology we used the FET to create a Legendre polynomial approximation for the distribution of scalar flux for a 2-D PWR pin cell. The numerical results confirm that the FET converges to the correct spatial flux distribution.

Point KENO V.a: A Continuous-Energy Monte Carlo Code for Transport Applications

Michael E. Dunn, N. Maurice Greene, and Lester M. Petrie

Oak Ridge National Laboratory, P. O. Box 2008, Oak Ridge, Tennessee 37831-6170, USA

KENO V.a is a multigroup Monte Carlo code that solves the Boltzmann transport equation and is used extensively in the criticality safety community to calculate the effective multiplication factor of systems with fissionable material. In this work, a continuous-energy or pointwise version of KENO V.a has been developed by first designing a new continuous-energy cross-section format and then by developing the appropriate Monte Carlo transport procedures to sample the new cross-section format. In order to generate pointwise cross sections for a test library, a series of cross-section processing modules were developed and used to process 50 ENDF/B-VI Release 7 nuclides for the test library. Once the cross-section processing procedures were in place, a continuous-energy version of KENO V.a was developed and tested by calculating 46 test cases that include critical and calculational benchmark problems. The Point KENO-calculated results for the test problems are in agreement with calculated results obtained with the multigroup version of KENO V.a and MCNP4C. Based on the calculated results with the prototypic cross-section library, a continuous-energy version of the KENO V.a code has been successfully developed and demonstrated for modeling systems with fissionable material.

Eigenfunction Convergence and Transmutation Enhancements in MCNPX

Gregg W. McKinney, John S. Hendricks, Holly R. Trellue, and Laurie S. Waters

Los Alamos National Laboratory, P.O. Box 1663, MS K575, Los Alamos, NM 87545 USA

This paper describes two new features developed for MCNPX. The first feature applies a variance reduction technique to achieve faster convergence of the eigenfunction in a criticality calculation. Results indicate that this new method converges nearly 100 times faster than the current approach. The second feature involves a transmutation option within MCNPX via a Fortran interface to the CINDER90 code. MCNPX burnup results for a simple criticality problem compare favorably to a Monteburns calculation.

Continuously Varying Material Properties and Tallies for Monte Carlo Calculations

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Using a high-order Legendre polynomial representation for material density and tallies within each cell, Monte Carlo codes can model continuous variations in material properties and results. We have demonstrated the Monte Carlo techniques for sampling the free-flight distances and performing pathlength flux tallies for this continuous representation. Application to both fixed-source and eigenvalue problems illustrates the benefits of the continuous representation as compared to conventional stepwise approximations. With these new methods, Monte Carlo codes can now be developed which are continuous in energy, angle, space, material properties, and tallied results.

Calculating the Effective Delayed Neutron Fraction Using Monte Carlo Techniques

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We present a true Monte Carlo estimator of the effective delayed neutron fraction β_{eff}. The advantage of this method is that by using the physics at the microscopic level, it obviates the need for adjoint calculations, without making any approximations. We have implemented this estimator into MCNP. In a standard k_{eff} calculation, the code now reports a β_{eff} value. The method does not slow down the code by more than 0.5%.

We validate our method with an extensive benchmark set. This set envelopes homogeneous and heterogeneous systems, fast and thermal spectra, as well as many different fuel types. The range of values for β_{eff} is 194–819 pcm. Among the 30 benchmark experiments are Masurca, FCA, TCA, Stacy, Proteus, and Godiva. We have performed calculations based on JEFF-3.0, ENDF/B-VI.8, and JENDL-3.3. Our method reproduces all experimental values.

We also compare with two approximate Monte Carlo methods, viz. the 'prompt' method and a method described recently by G.D. Spriggs et al. (Ann. Nucl. En. **28** (2001) 477). The 'prompt' method is reliable, but needs 40× more CPU time. The Spriggs method, on the other hand, needs only twice as much CPU time, but fails to reproduce the experimental values for heterogeneous systems.

Benchmarking of MONTEBURNS against Measurements on Irradiated UOX and MOX Fuels

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The analysis of fuel irradiated by MONTEBURNS showed that the simulation carried out using 40 JEF2.2 isotopes (main heavy isotopes and fission products) and 300 Endf-BVI isotopes (other actinides and fission products) for the 13 irradiated rods is very satisfactory. The average C/M variations for the fissile isotopes (U235, Pu239 and Pu241) is less than 3% and for the significant even isotopes (U236, Pu240, Pu242) less than 5%.

The maximum uncertainty (calculation + MONTEBURNS options + libraries+ experiment) is 5% for U235-U236 and 4% for the other heavy isotopes (1σ). The basic components of uncertainty are the burnup, the libraries and the thermal condition of the fuel rod.

A "fast" scheme with a significant MONTEBURNS time saving is possible within the framework of the sensitivity studies. In this case, one can retain a single ring for the fuel rod, 2 GWd/T for the burnup step, 10^{-3} for the importance

fraction and reduced MCNP4 convergence. Compared to the reference calculation, the time saving is of a factor of 20 approximately.

This work shows that MONTEBURNS is able to accurately model the depletion of UOX and MOX fuels up to 71.0 GWd/T and 47.2 GWd/T respectively. As the reference scheme depends on the continuous energy libraries used and on the exact representation of the real geometry only, it is also able to model any configuration representative of the future PWR, BWR, HTR and VHTR reactors with the same accuracy.

Analysis of Benchmark Results for Reactor Physics of LWR Next Generation Fuels

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Burnup calculation benchmark has been carried out for the LWR next generation fuels aiming at high burnup up to 70GWd/t with UO_2 and MOX. This paper summarizes the second term (FY2001-2002) activity of "Working Party on Reactor Physics for LWR Next Generation Fuels."

Based on the submitted results, the present status of calculation accuracy for the LWR next generation fuels has been confirmed, and the factors causing the calculation differences were analyzed in detail. Moreover, the future experiments and research subjects necessary to reduce the calculation differences were discussed and proposed.

Improvements of Isotopic Ratios Prediction through Takahama-3 Chemical Assays with the JEFF3.0 Nuclear Data Library

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This paper presents the interpretation of a destructive experiment of PWR fuels from the Takahama-3 reactor with the latest versions of the JEFF European nuclear data library. Such experiments, widely used in the JEF2.2 testing process, provide meaningful information to test specific cross-sections (or fission yields) in the thermal and resonance range. It is demonstrated in this work that the new JEFF3.0 library greatly improves main actinides and fission products isotopic ratio predictions compared with JEF2.2 results. The incorporation of JEFF3.0 evaluations within the multigroup nuclear library CEA2003V1 contributes towards the improvement of the APOLLO2 neutronic prediction capabilities. In particular, the new U235, U238 and Pu241 files adopted in JEFF3.0 remove longstanding discrepancies observed in U236, Np237 and Pu242 build-up prediction with JEF2.2 and reduce C/E discrepancies of minor actinides. Despite these improvements, further differential measurements and evaluation work are suggested to achieve a better accuracy in the prediction of Americium and Curium isotopes.

Analysis of the HTR-10 Initial Core with a Monte Carlo Code MVP

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The HTR-10 initial core has been analyzed by using a continuous-energy Monte Carlo code MVP with the nuclear data libraries JENDL-3.3, ENDF/B-6.8 and JEFF-3.0 to ascertain the accuracy of the code and address the bias of the libraries. The MVP statistical geometry model has been employed to treat the heterogeneity of stochastic materials such as randomly distributed pebbles and coated fuel particles. Although the statistical geometry model had a limitation that only one kind of statistically distributed particles can be treated in a region, the model has been improved to treat fuel and moderator pebbles in the HTR-10 core in this work. All the k_{eff} results agree with the experimental one within the deviation of about 0.8%. To be precise, the JENDL-3.3, ENDF/B-6.8 and JEFF-3.0 results overestimate by about 0.4%,

0.8% and 0.7%, respectively. The impact of nearest neighbor distributions (NNDs) has been also investigated for the k_{eff} values. No significant difference can be seen between the theoretical NNDs and the ones obtained by the MCRDF code.

OECD/NEA International Benchmark on 3-D VENUS-2 MOX Core Measurements

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For validating the calculation methods and nuclear data used for the prediction of power in MOX-fuelled systems, a series of theoretical physics benchmarks and multiple recycling issues of various MOX-fuelled systems have been addressed by the OECD/NEA. This led to many improvements and clarifications in nuclear data libraries and calculation methods. The final validation requires linking those findings to data from experiments. Hence, the first experiment-based benchmarks using the VENUS-2 MOX core measurement data have been started since 1999. The two-dimensional benchmark was completed in 2000. Overall, the results were very encouraging and confirmed that present methods using the latest nuclear data sets can adequately calculate MOX-fuelled systems. However, the calculation overestimated fission rates of MOX pins and slightly underestimated those of UO_2 pins. A full three-dimensional benchmark using 3-D VENUS-2 MOX core experimental data was therefore launched in 2001 for a more thorough investigation of the calculation methods. Twelve participants contributed to the 3-D benchmark, providing more than 20 solutions. This paper provides a summary of the comparison analysis of the 3-D calculation results against experimental data. Results obtained with the latest nuclear data libraries and various modern 3-D calculation methods are analyzed.

BN-600 Full MOX Core Benchmark Analysis

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As a follow-up of the BN-600 hybrid core benchmark, a full MOX core benchmark was performed within the framework of the IAEA co-ordinated research project. Discrepancies between the values of main reactivity coefficients obtained by the participants for the BN-600 full MOX core benchmark appear to be larger than those in the previous hybrid core benchmarks on traditional core configurations. This arises due to uncertainties in the proper modelling of the axial sodium plenum above the core. It was recognized that the sodium density coefficient strongly depends on the core model configuration of interest (hybrid core vs. fully MOX fuelled core with sodium plenum above the core) in conjunction with the calculation method (diffusion vs. transport theory).

The effects of the discrepancies revealed between the participants' results on the ULOF and UTOP transient behaviours of the BN-600 full MOX core were investigated in simplified transient analyses. Generally the diffusion approximation predicts more benign consequences for the ULOF accident but more hazardous ones for the UTOP accident when compared with the transport theory results. The heterogeneity effect does not have any significant effect on the simulation of the transient. The comparison of the transient analyses results concluded that the fuel Doppler coefficient and the sodium density coefficient are the two most important coefficients in understanding the ULOF transient behaviour. In particular, the uncertainty in evaluating the sodium density coefficient distribution has the largest impact on the description of reactor dynamics. This is because the maximum sodium temperature rise takes place at the top of the core and in the sodium plenum.

JOYO MK-III Performance Test at Low Power and Its Analysis

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JOYO is a Japanese experimental fast reactor whose missions include improvements in fast reactor safety and operation, and especially irradiation testing of advanced fuels and materials. JOYO was recently upgraded to the MK-III design, the second major upgrade since it began operation in 1977. MK-III performance test began in July 2003 to fully characterize the upgraded core and heat transfer system. Results of the test conducted at low power, which focus on the neutronics characteristics, and its analysis by a deterministic standard calculation method developed by Japan Nuclear Cycle Development Institute and the statistical calculation method are presented. These calculations were performed with JENDL-3.2.

Calculated values agree well with the experimental ones within 0.55 %Δk /kk' in the excess reactivity, 4% in the control rod worth and 3% in the isothermal temperature coefficient.

As a result, further investigation is required in the analysis. In the control rod worth, a discrepancy of C/E values between control rods in the third and the fifth rows are observed, especially for the half-traveled worth. In the excess reactivity, a difference between the results obtained by the deterministic method and the statistical method was about 0.5%Δk/kk'.

High Moderation Boiling Water Reactors Fully Loaded with MOX Fuel: The BASALA Experimental Program.

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This paper is devoted to the BASALA experimental program launched in the EOLE facility in CEA-Cadarache within the framework of a cooperation between the CEA, COGEMA and the Japanese organization NUPEC. This programme, performed between 2000 and 2002, was aimed at investigating the physical phenomena occurring in High Moderation Boiling Water Reactors fully loaded with MOX fuel. The first configurations (BASALA-H) were dedicated to simulate hot operating conditions by adjusting the moderation ratio and the second configurations (BASALA-C) to simulate further high moderation conditions than BASALA-H and corresponding to cold. One year was devoted for the experimental study for each phase in which reference and perturbed (2×8 and 2×16 Gd rods, voided and higher moderated cores, absorbing control blades) configurations were investigated. In addition integral soluble boron efficiency up to 600 ppm and the isothermal temperature coefficient up to 80°C were measured in the BASALA-C reference core. The first section presents the experimental programme and the main results. One shows that the experimental data were obtained with a very good accuracy, compatible with the required uncertainty to validate calculation tools for High Moderation BWR design. The second part describes the main calculation/experiment results obtained by CEA and NUPEC : the calculation reproduce rather well the experiment (i.e. the fission rate is given within ±1.5% discrepancy in average).

The Experimental Determination of the Relative Abundances and Decay Constants of Delayed Neutrons of the IPEN/MB-01 Reactor

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An in-pile experiment for the determination of the relative abundances and decay constants of delayed neutrons has been successfully performed at the IPEN/MB-01 research reactor facility. The experimental data are of good quality and can be used to validate theoretical predictions of the delayed neutron group constants based on the current knowledge of the fission products yields and emission probabilities for known precursors. The theory/experiment comparison shows that the current of ENDF/B-VI, namely release 8, shows severe discrepancies in both relative abundances and in the first decay constant. The revised version performed at LANL shows very good progress in both aspects. The best performance is obtained from JENDL3.3. One of the main achievements of the experiment was the consistency of the measured first decay constant to that of ^{87}Br. This consistency has never been proved in an in-pile experiment. Furthermore, for the first time it is shown experimental results from an in-pile experiment for an eight-group model.

The Experimental Determination of the Effective Delayed Neutron Parameters: β_{eff}, β_{eff}/Λ and Λ of the IPEN/MB-01 Reactor

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A reactor noise approach has been successfully performed at the IPEN/MB-01 research reactor facility in order to determine experimentally the delayed neutron parameters β_{eff}, β_{eff}/Λ and Λ. In the measurement of the β_{eff} parameter, the reactor power, which is of fundamental importance, was obtained with a very high degree of accuracy by a fuel rod scanning technique and a subsequent irradiation of highly enriched ^{235}U foil for the fission density normalization. The final measured values of β_{eff}, β_{eff}/Λ show very good agreement with independent measurements. The theory/experiment comparison shows deviations as large as 6.8% for β_{eff} when the ENDF/B-VI.8 library and its revised version performed at LANL are employed. For the β_{eff}/Λ, these deviations are of the order of 15.6%. The best agreement is obtained for JENDL3.3 library, where it is found a deviation of only 1.9% in the C/E ratio of β_{eff}. This result fully supports the reduction of the ^{235}U thermal yield as proposed by Okajima and Sakurai. The C/E ratio of β_{eff}/Λ is relatively higher, 11.1%, but this is due to the underestimation of the calculated prompt neutron generation time, Λ.

1D. Reactor Analysis Methods

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley). Session Chairs: Giovanni Bruna (Framatome ANP), Giuseppe Palmiotti (ANL).

Status of Reactor Analysis Methods and Codes in the U.S.A.

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The reactor physics methodologies and codes that are used in the U.S.A have been reviewed and summarized in this paper. The status of representative neutronics analysis capabilities and ongoing development activities are presented. This review covers cross-section generation capabilities for thermal and fast systems, whole-core deterministic (diffusion and transport) and Monte Carlo calculation tools, and depletion calculation methods and codes.

The review indicates that the existing neutronic analysis tools are sufficiently accurate for the design of current power reactors and for early pre-conceptual design development and viability phase evaluations of advanced reactor designs. For refined analyses of the advanced systems, however, they require additional verification and validation tests. Additional improvements and capabilities might be needed in order to reduce computational uncertainties and improve the plant operational economics.

Validation of WIMS9

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The WIMS lattice cell and burnup code has been established as a standard reactor physics code for a wide range of reactor types for the last 30 years, the latest version of which, WIMS8 [1], was issued in 1999. WIMS is continuously under development within Serco Assurance ANSWERS Software Service to meet the needs of its users and the increasing accuracy demands of the nuclear industry in general. As part of this development programme, a series of detailed studies were undertaken to compare the results from the deterministic WIMS modular code system with those from point data Monte Carlo calculations performed using the MONK8 code [2], a companion code in the ANSWERS code suite. Results of this investigation were outlined in a paper to the last PHYSOR conference [3]. These inter-comparisons led to the identification of the most significant method approximations remaining in WIMS8, in particular, in the resonance self shielding treatment. As a result of this, a number of improved methods have been developed and incorporated in WIMS for its next issue as WIMS9.

Application of the DSA Preconditioned GMRES Formalism to the Method of Characteristics – First Results

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The method of characteristics is well known for its slow convergence; consequently, as it is often done for discrete ordinates methods, the Generalized Minimal RESidual approach (GMRES) has been investigated for its practical implementation and its high reliability. GMRES is one of the most effective Krylov iterative methods to solve large linear systems. Moreover, the system has been left preconditioned with the Algebraic Collapsing Acceleration (ACA) a variant of the Diffusion Synthetic Acceleration (DSA) based on I. Suslov's former works.

This paper presents the first numerical results of these methods in 2D geometries with material discontinuities. Indeed, previous investigations have shown a degraded effectiveness of Diffusion Synthetic Accelerations with this kind of geometries. Results are presented for 9 x 9 Cartesian assemblies in terms of the speed of convergence of the inner iterations (fixed source) of the method of characteristics. It shows a signicant improvement on the convergence rate.

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley). Session Chairs: Giovanni Bruna (Framatome ANP), Giuseppe Palmiotti (ANL).

Improvement of the SPH Method for Multi-Assembly Calculations

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In this paper, improvement of the SPH method (the improved SPH method) is proposed. The SPH method is commonly used in pin-by-pin mesh core calculations to reduce cell-homogenization error. The investigation revealed that the normalization condition of the SPH factor in the conventional SPH method is not appropriate for multi-assembly calculations in which different assembly types are adjacent. Since the conventional normalization condition does not incorporate flux discontinuity between assemblies, cell homogenization error in assembly peripheral region becomes larger.

In the improved SPH method, the SPH factor is divided by an averaged "cell-level" discontinuity factor obtained in each fuel assembly. Though the SPH factor is somewhat changed from the conventional value, no additional homogenization parameters (e.g. discontinuity factor) is necessary in core calculations.

Test calculations were carried out in a simplified one-dimensional slab and two-dimensional PWR colorset geometries. The calculation results showed that the improved SPH method effectively reduce the cell-level homogenization error especially in assembly peripheral region.

Since we can easily implement the improved SPH method by slight code modifications, it can be a promising candidate of the cell-homogenization method for pin-by-pin core calculations.

The Characteristics Method Applied to a MTR Whole Core Modeling

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The Jules Horowitz Reactor (JHR) is the future European Material Testing Reactor dedicated to technological irradiations. These irradiations can be carried out thanks to several experimental devices placed in the core as well as in the reflector.

Due to its specific features a neutronics and thermalhydraulics simulation package named HORUS3D (Horowitz Reactor simulation Unified System) has been developed to model the JHR core. HORUS3D/N (which corresponds to the neutronics part) is based on the APOLLO2 and CRONOS2 codes.
In order to guarantee a high level of performance of this MTR, the margins have to be evaluated with a great confidence as well as the flux and the nuclear heating inside the experimental devices.

Therefore, the Nuclear Energy Division has developed a new calculation scheme based on a flux solver using the method of characteristics which has been recently implemented in the APOLLO2 transport code. The validation of this calculation scheme has been carried out by comparisons with reference Monte Carlo results. The core geometry is described in detail. The XMAS 172-group library with P3 scattering was applied and lead to an excellent result agreement with an acceptable computation time.

3D Characteristics Method with Linearly Anisotropic Scattering

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In this paper, we present the extension of the characteristics method to take into account anisotropic scattering effects in 3D geometries. Only the linearly anisotropic scattering cross sections and sources treatment will be considered here.

The method of characteristics is based on computing the angular flux following the tracking lines. Therefore, the implementation of anisotropic sources is relatively simpler than in collision probability method. However, the calculation of the anisotropic sources requires large amount of memory in order to store the angular flux per energy group, region, direction, and track. Comparisons between the isotropic and anisotropic fluxes, for different number of discretized angles and spatial meshing, are done. These results show that neglecting the anisotropic effects in scattering sources may induce significant errors on flux calculation especially on the regions near the reflected boundaries.

A Mutual Resonance Shielding Model Consistent with Ribon Subgroup Equations

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The paper describes the improvement of the lattice code component related to resonance self-shielding calculations. A new numerical scheme is proposed to represent the mutual shielding effect of overlapping resonances between different isotopes, in the context of the Ribon subgroup equations. The interference effects between two resonant isotopes are represented by a correlated weight matrix computed using a CALENDF approach. The model was designed with the primary goal of reducing the CPU ressources required to obtain the solution.

Finally, a validation is presented for the case of a mixture of ^{238}U and ^{240}Pu isotopes located in different geometries. The isotopic absorption rates obtained with the proposed numerical scheme are compared with exact values obtained using a fine-group elastic slowing-down calculation in the resolved energy domain.

Sensitivity Studies on Cross-section Generation and Modeling for BWR Core Simulation Using SAPHYR Code System

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The work presented in this report is a part of cooperative project between Pennsylvania State University and CEA-Saclay. The objective of the project is to establish an accurate and efficient scheme for BWR core calculation with coupled codes of the SAPHYR system and validate this methodology against the available experimental data. The project is divided into three stages and uses as reference design the Peach Bottom 2 plant. The Peach Bottom 2 plant was selected as a reference design because of the available measured data at different conditions, which allows us to validate both cycle and transient calculation capabilities of SAPHYR. In the first stage of the project, the main focus is on performing studies to implement and validate a cross-section generation and modeling methodology for boiling water reactor in order to identify and address the major sources of modeling assumptions in this process. The second stage involves studies on core-calculations with coupled 3D neutronics/thermal-hydraulic models and the relevant

issues to obtain an optimal representation in terms of accuracy and efficiency. The third stage will focus on the validation of the coupled calculation scheme against the available experimental data.

This paper presents the methodology for BWR assembly calculation with SAPHYR code system and shows results from performed sensitivity studies on cross-section modeling.

About Calculation of Axial Diffusion Coefficient in Nuclear Reactor Cells

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In this paper method for calculation of axial diffusion coefficient is developed. Purpose of this development is to analyze the significance of influence of neutron group spectrum of 2-dimensional problem into the value of axial leakage. To reach this purpose a way for calculation of direction probabilities in 2-D geometry has been suggested. Algorithm and code have been created on the base of this way. The code has been verified with the balance equation of the modeling problem.

The results obtained show that if the influence of heterogeneous structure of the neutron flux spectrum on the current along Z-axis is taken into account then for VVER-type cell the coefficient of axial diffusion is changed in fast group (~2%*)* and in thermal group (~15% This discrepancy can lead to the multiplication factor discrepancy about 200-300 pcm. This difference can influence on the neutron leakage in the area closed to reactor reflector. Code VEPS-A can be recommended to estimate the fine effects of axial diffusion coefficients calculations in reactor cell.

Uncertainty Analysis and Optimization Studies on the Deep-Burner Modular Helium Reactor (DB-MHR) for Actinide Incineration

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The MHR (Modular Helium Reactor) has been the subject of considerable design and analysis effort in the past. A variant of the concept, the Deep-Burner Modular Helium Reactor (DB-MHR), has been proposed by General Atomics to fit sustainability objectives.

It is characterized by a three-ring active core containing two different kinds of fuels:

- The Driver Fuel, consisting of the Plutonium and Neptunium discharged from LWRs,
- The Transmutation Fuel, consisting of the Minor Actinides also discharged from LWRs plus the Transuranics left in the DF after a complete irradiation cycle.

Transmutation worth is presently estimated at 65% for a critical operation at 1% $\frac{\delta k}{k}$., with an uncertainty of ±5%.

The paper investigates sources of such uncertainty:

- the cross-section libraries,
- the negative feed-back of graphite,
- other base-data, such as the branching rations in the Actinide chains,
- the burn-up fine spatial effects,
- the amount of energy released by fission,
- the stochastic effects on reactivity and power of micro-particle distribution inside the compacts,
- the computation options, such as the statistical precision of probabilistic calculations, the number of time-steps in burn-up calculation, the number of regions allowed to burn-out independently.

The paper also suggests ways to increasing the DB-MHR incinerating capacity by adjusting the core coupling, tuning of the neutron spectrum and modifying the fuel re-loading strategies.

PBMR Deep-Burn: A Pebble-Bed High Temperature Gas-Cooled Reactor Burning Its Own "Waste"

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A broader acceptance of nuclear power as a solution for the increasing energy demand will be directly linked to the solution of the radioactive waste issue. Lately a lot of effort has been dedicated on studying ways to minimise the amount of waste and to decrease the time span it should be stored safely in deep geological formations.

This work utilises the Deep Burn concept applied to the 268 MWth PBMR design to incinerate and transmute the long-lived transuranic elements in a single burn-up cycle. The reactor is fuelled partly with transmutation pebbles loaded with the transuranic elements originating from discharged UO_2 fuelled pebbles used in the same reactor. About 75% of the initial amount of transuranic waste is being incinerated, while the time span needed for the waste radiotoxicity to reach that of uranium ore is reduced to a third. The fissile component of the plutonium disappears almost completely (incineration of >99%). A total power of 58 MWth is produced by the transmutation pebbles, with a consequent 25% decrease in the amount of fresh UO_2 fuel needed.

A preliminary cost analysis shows advantages of operating the reactor in Deep Burn mode in countries with relatively high waste storage fees.

2A. Deep-Burn Physics and Methodology

Session Organizers: Alan Baxter (General Atomics), Giovanni Bruna (Framatome ANP), Alexander Stanceulescu (IAEA). Session Chair: Giovanni Bruna (Framatome ANP).

HTGR Actinide Burner Feasability Studies: Calculation Scheme Related Considerations

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For a few years, the High Temperature Gas cooled Reactor (HTGR) technology has gained worldwide new interest due to its specific characteristics. It is a promising reactor concept for the next generation of nuclear power applications. In addition to the studies performed on the industrial concept (uranium core type), there is a strong interest in the development of a «Deep Burner» version of HTGRs, dedicated to minor actinides destruction. At the CEA the actinides burner version of the prismatic block-type reactor is currently investigated, including studies about the design proposed by General Atomics. Advanced HTGRs allow a simplification of safety through reliance on innovative features and passive systems. One of the innovative features in HTGRs is the ceramic-coated fuel particle, which can retain the fission products even under severe accident conditions. The other original feature is the annular configuration of the core, surrounded by internal and external graphite reflectors able to evacuate decay heat in accidental conditions.

The growing interest in HTGRs activities has shown the need to validate the tools and techniques developed for reactor physics calculations. The physics of the deep-burn transmutation concept is still complicated due to the fact that two fuel types, driver and transmuter, are loaded in the core. Same ceramic-coated particle technology is used to fabricate both fuel types but particles are sized differently so as to favour immediate fissions in the driver fuel, or absorption followed by fissions in the transmutation fuel. Furthermore, self-shielding effects are of great importance to calculate HTGR fuel cycle or temperature coefficient correctly.

The purpose of this paper is essentially to evaluate the capability of the deterministic methods to calculate a wide range of core configurations. In the first part of the paper, the analysis is carried out on the «Deep Burner» fuel element geometry. In the second part, the analysis deals with the core geometry in order to estimate the impact of some physical assumptions on the fine fuel isotopic depletion.

Studies of a Deep Burn Fuel Cycle for the Incineration of Military Plutonium in the GT-MHR using the Monte-Carlo Burnup Code

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The graphite moderated and helium cooled reactors may play an important role in the future development of nuclear energy because of their unbeatable benefits: passive safety mechanism, low cost, flexibility in the choice of fuel, high conversion energy efficiency, high burnup, more resistant fuel cladding (because of the TRISO particles) and low power density. General Atomic possesses a long time experience with this type of reactors and it has recently developed a design in which this type of reactor is structured into 4 modules of 600 MW_{th}: the Gas Turbine – Modular Helium Reactor (GT-MHR). Since the GT-MHR offers a rather large flexibility in the choice of fuel type, Th, U, and Pu may be used in the manufacture of fuel with a quite ample degree of freedom. As a consequence, the GT-MHR may operate for very different purposes: e.g. the reduction of waste production trough fuel cycles based on thorium, which is quite attractive proposal for countries which approach for the first time the market of nuclear energy, the transmutation of LWRs waste or military Pu.

In the previous studies we analyzed the behavior of the GT-MHR with a fuel built on LWRs waste; whereas, in the present studies we tried to focus on the incineration of military Pu. This choice of fuel requires a detailed numerical modeling of the reactor, since the pretty high value of k_{eff} at the beginning of the fuel cycle does not allow to neglect the control rods and burnable poison in the computing simulations. By contrast, when the reactor is fueled with LWRs waste the breeding of fissile isotopes, at the equilibrium of the fuel composition, keeps almost constant and close to the criticality the value of k_{eff}.

Spectral Shift Methodology for “Deep-Burnup” of Uranium-Thorium-Hydride Fuel

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This paper examines the possibility of using spectral shift methodology within the reactor core made of thorium-hydride rods, by introducing hollow tubes inside the assembly. The spectral shift option is simulated by means of the one-dimensional lattice code WIMSD-5B through changing fuel-to-water volume ratios at different burnup steps. The analysis shows that it is possible to increase the discharge burnup values by 50% for uniform core configuration. The thorium-hydride fuel also possesses very good inherent safety features such as a large prompt fuel temperature coefficient of reactivity that is twice larger that of oxide fuel. This fuel also has very favorable non-proliferation characteristics.

Cascade Reactor Concept for Neutron Multiplication of Subcritical Core

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A “cascade reactor concept” represents a kind of system with an accelerator driven subcritical core consisting of two sections with different neutron spectra. It can generate more neutrons than one-section fast core with the same subcriticality and the same neutron source strength.

This study deals with the effect of the cascade reactor concept by analogy of a coupling core and direct simulation using a Monte Carlo code. The proton energy is assumed to be 1.0 GeV and a Pb-Bi target is at the center of the core. The cascade reactor has two sections. The inner section is a fast flux core and the outer section is a thermal flux core. There is a neutron absorber between two sections. As the result of direct simulation, the cascade reactor concept is confirmed to generate twice as many neutrons as a one-section reactor with the same neutron source. It is found that the thickness of fast core should be small to enhance the cascade effect, the thickness of Sm absorber is optimized to be 5 cm and that cascade effect increases when the subcriticality becomes small. These results are consistent with the basic theory of coupling core.

2B. Accelerator Applications and Spallation Physics

Session Organizers: Eric Pitcher (LANL), Stefano Monti (ENEA), Hiroyuki Oigawa (JAERI).
Session Chair: Eric Pitcher (LANL).

Research and Development Activities for Accelerator Driven System at JAERI

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The Japan Atomic Energy Research Institute (JAERI) is developing an Accelerator Driven System (ADS) for transmutation of nuclear waste such as minor actinide (MA) and long-lived fission product (LLFP). To study and evaluate the feasibility of ADS by physical and engineering viewpoints, the Transmutation Experimental Facility (TEF) is proposed under a framework of J-PARC (Japan Proton Accelerator Research Complex) project. The TEF consists of two facilities named as Transmutation Physics Experimental Facility (TEF-P) and ADS Target Test Facility (TEF-T). The TEF-P consists of a zero-power critical assembly which is operated with a low power proton beam to research the reactor physics and the controllability of ADS. The TEF-T is a facility for material irradiation and partial mockup of beam window which can accept a maximum 600MeV-200kW proton beam into the Pb-Bi eutectic target. The purposes, experimental items and the specifications of the facilities are described.

Research on Accelerator Driven Subcritical Reactor at Kyoto University Critical Assembly (KUCA) with the FFAG Proton Accelerator

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At Kyoto University Research Reactor Institute (KURRI), a new project for research on the accelerator driven subcritical reactor (ADSR) has been started form 2002. In this project, a new ring type accelerator based on the up-to-date FFAG (Fixed Field Alternating Gradient) technology is constructed where proton beam with arbitrary energy up to 150 MeV can be generated and the proton beam from this accelerator is introduced into a core of Kyoto University Critical Assembly (KUCA) to generate high energy neutron by collision with heavy metal such as tungsten. The accelerator system is now under construction and new experiments at KUCA combined with FFAG accelerator will be stared from 2005. Before starting this new experiment, basic research on ADSR has been performed at the KUCA combining with a accelerator to generate 14 MeV neutrons by D-T reaction.

The TRADE Experiment: Status of the Project and Physics of the Spallation Target

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The neutronic characteristics of the target-core system of the TRADE facility have been established and optimized for a reference proton energy of 140 MeV. Similar simulations have been repeated for two successive upgrades of the proton energy, 200 and 300 MeV, corresponding to different performances and design requirements and different characteristics of the proposed cyclotron and, as a consequence, of the proton beam. An extensive comparison of the main physical parameters has been also carried out, in order to evaluate advantages and disadvantages of different proton beam energies in the design of the spallation target and to allow the optimal engineering design of the whole TRADE facility.

2B. Accelerator Applications and Spallation Physics

Session Organizers: Eric Pitcher (LANL), Stefano Monti (ENEA), Hiroyuki Oigawa (JAERI).
Session Chair: Eric Pitcher (LANL).

Impact of Heterogeneous Cm-Distribution on Proton Source Efficiency in Accelerator-Driven Systems

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The proton source efficiency (ψ^*) was studied for homogeneous and heterogeneous distributions of minor actinides in a nitride-fuelled and lead-bismuth-cooled accelerator-driven system. The findings from the MCNPX simulations indicate that, compared to a homogeneous configuration, a gain in ψ^* by up to 16% can be obtained by distributing the minor actinides heterogeneously, Cm being placed in the inner zone of the active core and Am in the outer zone. The reason for this is the higher fission probability for neutrons for Cm than for Am in the energy range below 1.0 MeV.

Moreover, a comparative study of two different physics packages available in MCNPX, the Bertini and the CEM models, has been performed, focusing on the production of neutrons in the spallation target and on the proton source efficiency. The Bertini model was found to produce a higher number of neutrons in the low-energy range (below ~15 MeV) than the CEM model. Consequently, the Bertini model also over-estimates ψ^* by about 10%, compared to the CEM model.

Reactor-Accelerator Coupled Experiments (RACE): A Feasibility Study at TAMU

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A series of accelerator driven system (ADS) experiments are being planned to conduct demonstration and benchmark studies involving a nuclear reactor in a subcritical condition coupled to an accelerator driven neutron source. These experiments are being planned to use various levels of criticality and various power levels allowing for kinetics evaluation with and without temperature feedback. The reactor fuel to be used in these experiments is TRIGA reactor fuel. These reactor fuels are inherently safe and the reactor systems involved have large coolant capacities to ensure safe operation. The cores to be considered will be fully-instrumented and will allow for detailed and accurate data on core power levels, temperatures, and neutron fluxes during the experiments. A feasibility study is currently in progress at Texas A&M University (TAMU) to determine the viability and capability of these experiments. Preliminary results of this feasibility study are given.

Towards an Improved GELINA Neutron Target

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The neutron target of the GELINA time-of-flight facility has been simulated with coupled electron-photon-neutron MCNP4C3 calculations. The main goal is to investigate the feasibility for the reduction of the neutron time spread without significantly compromising the neutron yield. An implementation of new photonuclear libraries into the Monte Carlo calculations was necessary to simulate the neutronics problem accurately.

Absolute flux calculations were performed for both moderated and direct neutron spectra in order to validate the geometry model and the method used to deal with the neutronics problem. These results were compared with the measured neutron spectra at specific flight paths; an agreement within 20% was achieved. The resolution functions of the moderated spectrum currently in use were compared with our Monte Carlo simulations; here an excellent agreement was reached. Finally, the simulated resolution functions for the direct neutron spectrum were calculated with and without the presence of the moderator. This comparison illustrates the negative influence of the moderator on the resolution function for the fast neutron spectrum.

Displacement Kerma Cross Sections for Neutron Interactions in Molybdenum

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Atomic displacements are accepted as the principal underlying radiation damage mechanism for energetic neutron radiation in many materials. It is believed that accumulated displacements at the microscopic level form the basis for the changes in material properties at the macroscopic level. Therefore, it is important to quantify the amount of displacements caused to such materials in the radiation field corresponding to their intended use. If it is accepted that the total number of accumulated displacements determines the effect in the material, then displacements per atom (dpa) would provide a means for correlating damage in a given experimental testing field to damage in the expected operational field. The rational calculation of dpa requires the availability of displacement kerma cross sections for the materials of interest. The displacement kerma cross section has been a useful tool in calculating dpa because it allows the integration of the energy-dependent response of the material to the neutron radiation environment.

A modified version of the HEATR module in the NJOY99 code was developed and used to model non-monatomic solids more accurately by incorporating both the modified Kinchin-Pease (NRT) model of displacement damage and the partition of energy between emitted charged particles and recoil nucleus. The neutron cross section data used for the NJOY calculations were obtained from ENDF/B-VI. To account for the uncertainty in the displacement energy threshold, a range of plausible values that bracketed the range of accepted values was used for the calculations. The effects of the displacement threshold energy variations are most obvious in the energy ranges 1-10 keV and below 10 eV. Differences between the displacement kerma cross sections for the extreme values of E_d in the thermal range are approximately 20%.

Thermal Neutron Scattering Cross Sections of Thorium Hydride

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An ab initio approach is used to calculate the phonon frequency distribution of thorium hydride (ThH_2). The scattering law and the thermal neutron scattering cross sections for Th and H have been generated using this distribution. Due to the lack of experimental data, the results have been compared to zirconium hydride (ZrH_2) data that were generated using a classical lattice dynamics model. Based on this comparison, it is shown that the differences in properties of ThH_2 and ZrH_2 are entirely consistent with the physical differences between the two materials. Consequently, this work provides a set of data that is useful in nuclear reactor technology, and illustrates the utility of the ab initio approach as a unique predictive tool of the properties of materials that are of interest to nuclear reactor design.

Three-Dimensional RAMA Fluence Methodology Benchmarking

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This paper describes the benchmarking of the RAMA Fluence Methodology software that has been performed in accordance with U. S. Nuclear Regulatory Commission Regulatory Guide 1.190. The RAMA Fluence Methodology has been developed by TransWare Enterprises Inc. through funding provided by the Electric Power Research Institute, Inc. (EPRI) and the Boiling Water Reactor Vessel and Internals Project (BWRVIP). The purpose of the software is to provide an accurate method for calculating neutron fluence in BWR pressure vessels and internal components. The Methodology incorporates a three-dimensional deterministic transport solution with flexible arbitrary geometry representation of reactor system components, previously available only with Monte Carlo solution techniques. Benchmarking was performed on measurements obtained from three standard benchmark problems which include the Pool Criticality Assembly (PCA), VENUS-3, and H. B. Robinson Unit 2 benchmarks, and on flux wire measurements obtained from two BWR nuclear plants. The calculated to measured (C/M) ratios range from 0.93 to 1.04 demonstrating the accuracy of the RAMA Fluence Methodology in predicting neutron flux, fluence, and dosimetry activation.

Ab Initio Generation of Thermal Neutron Scattering Cross Sections

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Quantum mechanical ab initio (i.e., first principle) methods are applied in generating the thermal neutron scattering cross sections of moderators and reflectors that are of interest in nuclear technology. Specifically, this work focuses on graphite and beryllium. In both cases, the ab initio code VASP and the lattice dynamics code PHONON were used to generate the dispersion relations, and the phonon frequency distributions (density of states). This information was then utilized in the LEAPR module of the NJOY code to calculate the thermal neutron scattering cross sections at various temperatures. The use of the ab initio approach represents a major departure from previously applied methods, which depended mainly on fitting simpler dynamical models to experimental data to arrive at the phonon frequency distributions. In this case, much more complicated models of the atomic system of interest can be set up, which allows the establishment of a more complete dynamical matrix. As opposed to the semi-empirical methods used previously, this method represents a fundamental and predictive approach for estimating materials' properties including ones that are of interest in nuclear reactor design.

Support Vector Machine in Classification of Positron Lifetime Spectra

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Positron annihilation spectroscopy is used in the study of radiation induced defects in nuclear materials in a non-intrusive way. The positron can be trapped by defects and the number of exponential components in the positron lifetime spectrum is related to the number of defect states. This work concerns classification of positron spectra with respect to the number of spectral components by support vector machine (SVM). The SVM has not yet been investigated for positron spectra analysis. The SVMs use an optimized generalization. The SVM classifier has been constructed by training with the simulated positron spectral data with two and three spectral components. Tuning the hyperparameters, such as the generalization parameter C, has been done by using the 10-fold cross-validation error. The experimental spectra available from polymer materials have been analysed by the constructed SVM nonlinear classifier. Experimental data are classified as the class of positron spectra with three spectral components with accuracy of 95.4 %. The SVM calculations show that certain degree of misclassification tolerance can produce a solution with good expected generalisation.

Research Reactor Application to Iridium-192 Production for Cancer Treatment

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The purpose of this work is to settle a laboratory for Iridium –192 sources production, that is, to determine a wire activation method and to build a hot cell for the wires manipulation, quality control and packaging. The paper relates, mainly, the wire activation method and it quality control. The wire activation is carried out at IEA- R1m nuclear reactor.

Session Organizer: Marvin Adams (Texas A&M University).
Session Chairs: Marvin Adams (Texas A&M University), Dmitriy Anistratov (North Carolina State University).

Renormalized Treatment of the Double Heterogeneity with the Method of Characteristics

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A numerical solution for the method of characteristics in structured meshes (MOC) is proposed to treat media comprising stochastic regions composed of a homogeneous matrix with a statistical dispersion of spherical grains of different types. The method incorporates directly an asymptotic stochastic solution in the MOC formalism without affecting the overall numerical performance and can be properly accelerated by the same MOC acceleration techniques. As opposed to an earlier approach, the new method uses the stochastic solution predicted by renewal theory to compute region transmission, and a normalizing coefficient is applied then to ensure conservation. Numerical comparisons are presented for a set of Kritzs experiments for MOX fuel, and for a 2D HTGR reactor with prismatic fuel elements. For the HTGR case the comparisons with a Monte Carlo calculation show that the renormalized MOC treatment predicts assembly powers within 1%.

Coarse-Mesh Discretized Low-Order Quasidiffusion Equations for Subregion Averaged Scalar Fluxes

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In this paper we develop homogenization procedure and discretization for the low-order quasidiffusion equations on coarse grids for core-level reactor calculations. The system of discretized equations of the proposed method is formulated in terms of the subregion averaged group scalar fluxes. The coarse-mesh solution is consistent with a given fine-mesh discretization of the transport equation in the sense that it preserves a set of average values of the fine-mesh transport scalar flux oversubregions of coarse-mesh cells as well as the surface currents, and eigenvalue. The developed method generates numerical solution that mimics the large-scale behavior of the transport solution within assemblies.

Application of a Heterogeneous Coarse Mesh Transport Method to a MOX Benchmark Problem

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Recently, a coarse mesh transport method was extended to 2-D geometry by coupling Monte Carlo response function calculations to deterministic sweeps for converging the partial currents on the coarse mesh boundaries. More extensive testing of the new method has been performed with the previously published continuous energy benchmark problem, as well as the multigroup C5G7 MOX problem. The effect of the partial current representation in space, for the MOX problem, and in space and energy, for the smaller problem, on the accuracy of the results is the focus of this paper. For the MOX problem, accurate results were obtained with the assumption that the partial currents are piecewise-constant on four spatial segments per coarse mesh interface. Specifically, the errors in the system multiplication factor and the average absolute pin power were 0.12% and 0.68%, respectively. The root mean square and the mean relative pin power errors were 1.15% and 0.56%, respectively.

2D. Advances in LWR Analyses

Session Organizer: Marvin Adams (Texas A&M University).
Session Chairs: Marvin Adams (Texas A&M University), Dmitriy Anistratov (North Carolina State University).

Spatially Consistent Coarse-Mesh Discretization of the Low-Order Quasidiffusion Equations in 2D Geometry

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In this paper, we develop spatially consistent coarse-mesh discretization method for the 2D low-order quasidiffusion equations for the full-core calculations. The coarse-mesh solution generated by this method preserves a number of spatial polynomial moments of the fine-mesh transport solution over coarse cells. The proposed method reproduces accurately the complicated large-scale behavior of the transport solution within assemblies. To demonstrate accuracy of the developed method, we present numerical results of calculations of test problems that simulate interaction of MOX and uranium assemblies.

The Use of an Artificial Neural Network for On-Line Prediction of Pin-Cell Discontinuity Factors in PARCS

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During the last several years, the U.S. NRC has sponsored the development of multi-group, SP_3 method in the Purdue Advanced Reactor Core Simulator (PARCS) primarily for MOX fuel analysis. These methods were implemented within the framework of pin-by-pin discretization using pin homogenized cross sections. Pin-Cell Discontinuity Factors (PDFs) were proposed in order to recover the error introduced by pin cell homogenization. The method showed the potential to improve the accuracy of the pin power prediction; however its application for practical core problems was limited because of the considerable amount of data required for whole core calculations and because of uncertainties in the application of PDFs to core conditions for which they were not generated. The work reported here is the development of innovative methods to implement PDFs for practical applications using an Artificial Neural Network (ANN). The work is demonstrated using the KAIST MOX benchmark.

Developing a Basis for Predicting and Assessing Trends in BWR Core Tracking

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In the process of designing a boiling water reactor (BWR) fuel cycle, it is necessary to estimate the bias eigenvalue trend or nuclear design basis (NDB), which has a large effect on cycle parameters. Currently, history of previous cycles is used as a basis, but due to constant demands of higher energy output per cycle, and unexpected events during a cycle, predictions become challenging. In addition to known variations, there may also be unrecognized events that may cause the eigenvalue and other plant parameters to vary. Considering that safety and cost can be greatly affected by incorrect predictions, it is important to understand the NDB trends when doing calculations for a future cycle, or evaluating eigenvalue drift for a current cycle. To aid in developing a basis for making predictions, various perturbations in the areas of fuel manufacturing and plant measurement were studied in a multicycle analysis. These perturbations have a range of effects on several different cycle parameters. The results are intended to assist in the prediction and assessment of trends in the BWR industry.

New Resonant Mixture Self-Shielding Treatment in the Code APOLLO2

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Up to now, the self-shielding module of the multigroup transport code APOLLO2 could treat one resonant isotope mixed with moderator isotopes. Consequently, the resonant mixture self-shielding treatment was an iterative one. Each resonant isotope of the mixture was treated separately, the other resonant isotopes of the mixture being then considered as moderator isotopes, that is to say non-resonant isotopes. This treatment could be iterated. This approximate method could lead to discrepancies (versus reference calculations) of several percents on the resonant absorption reaction rates.

The new method consists in treating the resonant mixture as a unique entity. New developments were carried out to take into account (with the Livolant-Jeanpierre approach) this global entity, to dynamically recalculate in the code APOLLO2 the self-shielding data of the treated mixture and then to perform separately the self-shielded cross-sections of each component of the mixture.

This new treatment allows us to reduce the local discrepancies on resonant absorption reaction rates versus reference calculations. However, as this new method suppresses compensations, some global results such as effective multiplication factor can be slightly deteriorated.

Extension of KIN3D, a Kinetics Capability of VARIANT, for Modeling Fast Transients in Accelerator Driven Systems

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The time-dependent second-order PN equation (even-parity transport equation) can be discretized in time by a fully implicit finite-difference scheme that results in a set of pseudo steady-state PN equations solved at each time step by the Variational Nodal Method. For this procedure we consider employment of two techniques corresponding - for the lowest angular approximation - to (1) the transient diffusion and (2) to the P1/telegraph equation. The first scheme was implemented originally in the VARIANT/KIN3D code; the second one was implemented there recently. In the paper, the new method is presented and validated in P1 for a benchmark case by comparing with a result based on the analytical technique. In addition, a simplified experimental model is analyzed in order to get an impression: how the results obtained by the new technique may differ from those obtained by the original KIN3D version earlier. As expected, the results don't show substantial changes in the detector response to an external source pulse. On the other hand, the effect of employing the new option is not negligible: it may shift the results by several percents.

Reactivity Assessment and Spatial Time-Effects from the MUSE Kinetics Experiments

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The MUSE program is a series of zero-power experiments carried out at the CEA-Cadarache MASURCA facility since 1995 in the 5th European Framework, to study the neutronics of Accelerator Driven Systems (ADS). Among the main purposes of the MUSE experimental program there is the analysis of the possibility to infer the subcritical level of a source driven system using Pulse Neutron Source (PNS) methods, in view of the extrapolation of such methods to an European Transmutation Demonstrator (ETD).

In this aim, one of the MUSE-4 phase objectives is the investigation of the system response to neutron pulses, provided by the GENEPI pulsed deuteron accelerator, with frequencies from 50 Hz to 4.5 kHz, and less than 1μs wide, generated at the reactor center by (d, d) and (d, t) reactions. Detectors mainly based on ^{235}U fission, located in the core, reflector and shielding regions are used for measuring the time dependent responses.

In this paper, the results obtained for a MUSE-4 configuration characterized by a low subcritical level (about -12 $) have been analyzed on the basis of both theoretical backgrounds and corresponding calculation simulations. In particular, both PNS area and α-fitting classical methods for measuring the subcriticality level have been analyzed, giving indications about the performance of such methods when applied to the MUSE case.

Monte Carlo Modeling of a Time-of-Flight (TOF) Experiment for Determination of Fe Scattering Cross Sections

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This paper discusses the Monte Carlo modeling of a time-of-flight (TOF) experiment used to investigate the Fe-56 neutron scattering cross section. This involves utilizing experimental data to provide a Monte Carlo neutron source distribution for which experiments can be compared with simulation. Results indicate this can be an effective methodology for the generation of a source of this kind. Comparisons of calculation and experiment of the time of flight experiment have shown possible deficiencies in the Fe-56 scattering cross section.

Application of the Dynamic Control Rod Reactivity Measurement Method to Korea Standard Nuclear Power Plants

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To measure and validate the worth of control bank or shutdown bank, the dynamic control rod reactivity measurement (DCRM) technique has been developed and applied to six cases of Low Power Physics Tests of PWRs including Korea Standard Nuclear Power plant (KSNP) based on the CE System 80+ NSSS. Through the DORT results for each two ex-ore detector response and the three dimensional core transient simulations for rod movements, the key parameters of DCRM method are determined to implement into the Direct Digital Reactivity Computer System (DDRCS). A total of

9 bank worths of two KSNP plants were measured to compare with the worths of the conventional rod worth measurement method. The results show that the average error of DCRM method is nearly the same as the conventional Rod Swap and Boron Dilution Method but lower standard deviation. It takes about twenty minutes from the beginning of rod movement to final estimation of the integral static worth of a control bank.

Physics Characteristics of U-$ZrH_{1.6}$ Fueled PWR Cores

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This paper summarizes part of the neutronic studies performed for NERI project 02-189 aimed at assessing the feasibility of improving the performance of light water reactors by using solid hydride fuel instead of oxide fuel. Infinite lattice neutronic parametric study indicates that it should be possible to design PWR cores using U-$ZrH_{1.6}$ fuel to have comparable discharge burnup as attainable with UO_2 fuel of same enrichment. The optimal hydride fuel P/D ratio is significantly smaller than that of oxide fuel but has a softer spectrum. U-$ZrH_{1.6}$ fuelled cores can be designed to have negative coolant and fuel temperature coefficient of reactivity. U-$ZrH_{1.6}$ fuel offers a couple of unique temperature reactivity feedback mechanisms – a prompt feedback due to spectrum hardening and a delayed feedback due to hydrogen diffusion. Handling of hydride fuel will require special attention to criticality safety.

CANDU Adjuster Rods Incremental Cross Section Evaluation: A Perturbative Approach

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The effects of the main reactivity control devices in finite CANDU reactor calculations are generally taken into account via incremental cross sections. These are obtained by comparing the cell averaged cross sections resulting from two successive 3-D transport calculations, namely a first supercell calculation with the control devices inserted and a second calculation with the control devices extracted. Computing such incremental cross sections therefore represents a good candidate for the application of a generalized perturbation theory (GPT) method. Such a method has been recently implemented in DRAGON for collision probability (CP) based transport calculations. It will be used to evaluate the incremental cross sections associated with CANDU adjuster rods.

Using the unrodded supercell as a reference, we show that perturbation theory results are very good for the incremental cross section associated with the guide tube. However, relatively large errors are observed for the adjuster rods themselves. Selecting as the reference for our perturbative evaluation the supercell with adjuster 1 inserted yields incremental cross sections for the remaining 5 adjusters. The results generally lies within the precision limit expected from the exact evaluation. Using this approach represents a gain of a factor of 4 in computation time as compared with the exact evaluation. This is appreciable for current CANDU applications and very promising for the design of new reactors concepts where the exact composition and/or dimension of the control rods are not yet finalized.

3B. Research Reactors

Session Organizers: Nelson Hanan (ANL), Hamid Ait Abderrahim (SCK-CEN).
Session Chairs: R. Trenton Primm III (ORNL), Thomas H. Newton Jr. (MIT).

Modeling the MIT Reactor Neutronics for LEU Conversion Studies

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To investigate the possible use of high-density low enriched uranium (LEU) in the MIT Reactor, the core neutronic behavior has been modeled using the Monte Carlo based codes MCNP for statics and MCODE for burnup. These models have been validated using criticality, blade worth, and burnup reactivity measurement data from MITR-II Core #2. Preliminary studies indicate that the LEU novel Mo-U fuel U7Mo can provide the reactivity needed to operate the reactor for adequate life times. The LEU core design optimization to maximize the neutron flux is proceeding using these models.

A New De-Homogenisation Method for Local Power Reconstruction

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The transport calculation of a MTR core takes too much computation time for design studies and approximations have to be introduced: diffusion theory is used and the assemblies are homogenized. Nevertheless, safety studies require the knowledge of the local power of the fuel plates. The standard reconstruction method assumes the detailed flux shape in an assembly as a superposition of a macroscopic flux in the core and a microscopic one at the level of the assembly. Because of the strong flux gradients, the MTR core-reflector interface is obviously the most difficult point for the reconstruction method. A constant flux hypothesis in the assembly leads there to discrepancies which may rise by up to 22% on the Horowitz reactor core.

We have developed an original self-adjusting piecewise P_1/P_3t polynomial approximation of the "macroscopic" flux over each assembly of the JHR core which is thereafter multiplied classically by local flux distribution.
The discrepancy against Monte-Carlo simulation reaches so only 6.5% on the power peak.

Reactor Physics Studies of Reduced-Tantalum-Content Control and Safety Elements for the High Flux Isotope Reactor

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Some of the unirradiated High Flux Isotope Reactor (HFIR) control elements discharged during the late 1990s were observed to have cladding damage—local swelling or blistering. The cladding damage was limited to the tantalum/europium interface of the element and is thought to result from interaction of hydrogen and europium to form a compound of lower density than europium oxide, thus leading to a "blistering" of the control plate cladding. Reducing the tantalum loading in the control plates should help preclude this phenomenon. The impact of the change to the control plates on the operation of the reactor was assessed.

Regarding nominal, steady-state reactor operation, the impact of the change in the power distribution in the core due to reduced tantalum content was calculated and found to be insignificant. The magnitude and impact of the change in differential control element worth was calculated, and the differential worths of reduced tantalum elements vs. the current elements from equivalent-burnup *critical* configurations were determined to be unchanged within the accuracy of the computational method and relevant experimental measurements. The location of the critical control elements symmetric positions for reduced tantalum elements was found to be 1/3 in. less withdrawn relative to existing control elements regardless of the value of fuel cycle burnup (time in the fuel cycle). The magnitude and impact of the change

in the shutdown margin (integral rod worth) was assessed and found to be unchanged. Differential safety element worth values for the reduced-tantalum-content elements were calculated for postulated accident conditions and were found to be greater than values currently assumed in HFIR safety analyses.

Automated Three Dimensional Depletion Capability for the Pennsylvania State University Research Reactor

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This paper describes the new depletion calculation capability for the Pennsylvania State University research reactor (PSBR). The calculation scheme employs the general purpose Monte Carlo code MCNP4C coupled with the versatile nuclear depletion code ORIGEN2.2 via the recently developed interface code, TRIGSIM. The main functions of the TRIGSIM code are automatic generation of MCNP and ORIGEN inputs from one TRIGSIM input and interfacing the data exchange between the two codes. The PSBR core loading 1, core loading 2 and core loading 3 were modeled using TRIGSIM code. The excess reactivity results from the core calculations have shown reasonable agreement compared to the measured excess reactivity from the operational log book.

HOR: Criticality Comparison Using a Nodal Code, Monte Carlo Codes and Plant Data

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The core neutronic simulation of research reactor core reloads and cycles histories remains a challenging task, especially since the reactors are typically small and non-uniform in layout.

At the Interfaculty Reactor Institute (IRI) of the Delft University of Technology the HOR pool-type research reactor has been in a transition from high-enriched uranium (HEU) fuel to low-enriched uranium (LEU) fuel elements since 1998. At the same time the layout has been changing to a more compact core and some of the in-core and beam line configurations have changed.

In support of the transition from HEU to LEU, core reactor calculations and reactor physics experiments and measurements are continuously performed and evaluated to ensure safe and optimal operation. During the transition and part of normal reactor operations the reactivity, flux distributions, power distributions, burn-up distributions, and control rod reactivity were measured and calculated. Monte Carlo codes and a nodal code are used for these calculations. There is a good agreement between the calculations and plant data. In addition feasibility studies have been performed for modifying and upgrading the HOR with the aim to improve the utilisation performance in combination with the installation of a cold neutron source. The calculations suggested that such an ultra-compact core (3x3 elements) with high fuel loadings is possible.

3B. Research Reactors

Session Organizers: Nelson Hanan (ANL), Hamid Ait Abderrahim (SCK-CEN).
Session Chairs: R. Trenton Primm III (ORNL), Thomas H. Newton Jr. (MIT).

The Application of the Zirconium to the Thermal Neutron Fluence Monitoring at the Irradiation Experiments of the Research Reactor

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A determination procedure for the thermal neutron flux using zirconium is proposed. The zirconium foils and cobalt wires are irradiated simultaneously at the irradiation hole of HANARO in order to measure the thermal neutron flux. It is confirmed that the handling of a zirconium sample is much more convenient than that of the cobalt wire in the processes of sample preparation, irradiation and sample fixing for gamma-ray detection. It is confirmed that the two gamma-ray peaks at 724 and 757 keV from Zr-95 are useful for the flux measurements in the case of an irradiation for several hours at the neutron flux of ~3×10^{13} n/cm^{2}s. The thermal neutron flux measured within 8 days from the zirconium irradiation are slightly smaller than that after 8 days because of the effects of a high count rate and the interference of the gamma-rays from Nb-97. The neutron fluxes measured by zirconium are 15.2~16.7% larger than those by cobalt. Anyhow, the ratio of the flux measured by the cobalt to that by the zirconium can be determined. Therefore, the equivalent thermal neutron fluence for cobalt can be obtained by using the neutron flux determined by zirconium and the flux ratio. The uncertainty in the neutron flux determination is a little larger than that determined by cobalt only. However, it is expected that the above procedure is very useful when a number of monitors are needed in the irradiation experiments of the research reactors.

Determination of the Linear Power in MOX Fuel Rods Irradiated at the BR2 Reactor

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The high flux materials testing reactor BR2 is regularly used for the irradiation of new types of fuel elements. In particular the CALLISTO in-pile loop provides realistic PWR conditions for the irradiation of LWR fuel and/or materials. The loop comprises three in-pile sections (IPS); up to nine fuel rods can be loaded in each IPS.

The MCNP model of the high flux materials testing reactor BR2 was used for a simulation of the irradiation of MOX fuel rods. Detailed inter-comparison of experimental and theoretical methods of determining the distribution of power in MOX fuel rods is presented in this paper. Calculations of the effective heating energy per fission in the irradiated MOX fuel rods were performed in order to determine the absolute values of the thermal power in MOX rods. The gamma-spectrometric measurements as well as the results from the thermal balance method are compared with the theoretical calculations. The calculated linear power distribution in MOX rods differs from the measured distribution on average not more than 5%.

3C. Fuel Cycle Physics

Session Organizers: Alain Zaetta (CEA Cadarache), Michael Todosow (BNL).
Session Chairs: Michael Todosow (BNL), John Stillman (ANL).

The Physics of TRU Transmutation – A Systematic Approach to the Intercomparison of Systems

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In this collaborative effort, a methodology is developed to enable systematic analysis and comparison of diverse nuclear fuel cycle strategies. First, transmutation potential is assessed by considering the neutron balance for destruction of each actinide isotope; a range of thermal and fast reactor systems are considered. In general, a harder neutron energy spectrum results in a more favorable neutron balance. The method is extended to compute equilibrium actinide compositions for a generalized fuel cycle model (open or closed). In this paper, the technique is employed to compare the transmutation performance of 1) PWRs with varying moderator-to-fuel ratio, 2) PWR closed cycle strategies with varying treatment of the minor actinide elements, and 3) fast reactors with either plutonium or transuranic recycle. The method is demonstrated to quickly evaluate the main characteristics of the associated fuel cycles (e.g., isotopic mass flows, neutron balance for critical enrichment) and give some indication of other performance parameters (e.g., reactivity effects).

Dynamic Analysis of the AFCI Scenarios

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This work describes the dynamic analysis of different Advanced Fuel Cycle Initiative (AFCI) scenarios that aim at exploring the potential of using advanced nuclear energy systems to reduce the difficulties associated with the disposition of nuclear spent fuel. Different reactor deployment strategies have been considered including once-thru, single-recycle, and multiple-recycling scenarios, and its combinations. The impact of those strategies on repository performance has been investigated through the estimation of the waste heat load associated with each scenario. Conclusions regarding the impact of the different scenarios are presented.

Uncertainty Analysis on Back-End Fuel Cycle Main Parameters

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The present technology of nuclear energy production originates a long-term toxicity, which is roughly proportional to the mass of spent fuel. Plutonium is the most abundant of the man-made elements in spent fuel, it becomes, by the way, the source of the middle term activity. Transuranics dominate the long-term toxicity, which is significantly altered by decay.

A simplified approach was defined to compute the PWR fuel-mass vector of both UOX and MOX fuels within French standards, at the reactor discharge and at any intermediate time up to one-million years. It is based on linearization of fuel-life equations through a matrix operator. Fuel compositions at the core discharge are obtained applying the operator to the fresh fuel vector.

Uncertainties come from the linearity versus burn-up assumed in building-up operator, power history and heterogeneity of burn-ups at the discharge.

Fuel compositions at any time are obtained by applying the decay operator to the fuel vector at the core discharge. Although the approach is analytic, the neutron source in spent fuel and the propagation of errors affecting base-data generate uncertainties.

UOX and MOX fuels were investigated independently for enrichments in the range of the French current fabrication standards. The error on back-end parameters remains lower than 20%.

The paper summarizes the analysis, focusing on the main aspects, as the reliability of the procedure adopted to generate the matrixes, the impact of reactor power history and the uncertainties on decay data.

Effectiveness of Different Burnable Poisons in a Long Cycle BWR

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BWR fuel assemblies for very long cycle operation are investigated with three different types of burnable absorbers; gadolinium, erbium, and the B_4C in alumina annular burnable absorber. Their nuclear characteristics such as reactivity and power distributions are compared using CASMO-4/SIMULATE-3. The results show that the enriched Gd-157 gadolinium case has advantage in terms of lowest reactivity swing throughout the cycle. However, the local peaking factor (LPF) in the assembly calculation and the nodal peaking in the core calculation at beginning-of-life (BOL) were high. The erbium case showed more reactivity swing but the LPF and nodal peaking were lowest of all three cases. The B_4C case had the highest reactivity at BOL which must be suppressed by control rods, but the nodal peaking for B_4C was lower than the gadolinium case. The most important advantage of B_4C over others was the saving of uranium inventory needed to achieve the equivalent target exposure of 15EFPY. Further analysis for transient conditions must be performed to ensure meeting all transient limits.

Studies of Advanced Fuel Cycles in Indian Pressurized Heavy Water Reactors and Advanced Heavy Water Reactor

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In order to conserve the natural uranium resources, a study has been performed to use the depleted uranium and plutonium discharged from the PHWRs. A new fuel cluster design, MOX-888, has been proposed which contains 0.8 wt % PuO2 mixed with depleted uranium having 0.25 wt % U^{235}. It has been found that it is possible to use MOX-888 fuel clusters in the outer 190 channels in conjunction with natural UO_2 bundles in the central 116 channels of the reactor without making any changes in the other hardware of the reactor. The average discharge burnup of fuel (Natural UO_2 and MOX-888) can be improved to more than 10000 MWD/T from the present value of 6700 MWD/T resulting in the substantial saving of natural uranium per year. The paper discusses the physics studies pertaining to this fuel cluster. It also compares the worth of various control devices including primary and secondary shut down systems for the proposed core with present core having all natural uranium fuel clusters.

An Advanced Heavy Water Reactor is being designed making use of thorium fuel. The design envisages the recycling of ^{233}U produced due to capture of neutrons in the thorium. ^{234}U is also produced in this process. The paper discusses the results of multicycle studies performed to estimate the increase in the concentration of ^{234}U and associated burnup penalty in the reprocessed uranium in each recycle.

The paper also compares the production of minor actinides produced in the discharged fuel, their associated radioactivity and toxicity for PHWR, AHWR and PWR fuels. Comparison shows that the production of minor actinides per unit energy is less in AHWR fuel than those of PWR fuel, but it is comparable to the case of PHWR fuel. This is mainly due to presence of (Th-Pu) MOX pins in the AHWR fuel cluster.

Influence of Nuclear Fuel Cycle Duration and Reprocessing Losses Level on the Nuclear Power System Structure

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It is shown that three-component Nuclear Power (NP) system consisting of thermal reactors (TR), fast reactors (FR) and molten salt reactors-burners (MSR) can operate in mode when actinides in system are not stored up proportionally to energy generated but are practically at steady level proportionally to system power.

In the paper there was considered the influence of nuclear fuel cycle (NFC) duration and level of irretrievable reprocessing losses on NP structure and fuel consumption efficiency.

U-Pu and U-Th variants of NP systems were considered with various levels of irretrievable reprocessing losses for all actinides - 0%, 0.1% и 1%, and various duration of cooling and reprocessing time of TR fuel cycle (TRR time) – 3 years, 6 years, 9 years, 20 years. Steady state calculations were performed for three-component NP systems and there were obtained steady state amounts and radioactivity of actinides for main components of NFC closed by actinides. Also, there were obtained rates and radioactivity of accumulation of actinide irretrievable reprocessing losses.

Assessment of Reduced Moderation Water Reactor Fuel Cycle

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An assessment of the fuel cycle performance of the Reduced-Moderation Water Reactor (RMWR) concept has been performed. The transuranics consumption rate, key safety coefficients, fuel handling issues, natural uranium requirements, and radiotoxicity of the waste passed to the repository or interim storage were evaluated and compared to other reactor-based transmutation options.

The decay heat and dose rate of the RMWR fuel are about 2 to 3 times higher than the corresponding values of the multirecycling LWR fuel (CORAIL concept) at the fabrication and charge stages because of the high plutonium content of the RMWR fuel. The radiotoxicity per Giga-Watt-day electric of the RMWR waste is smaller than that of the CORAIL waste 10 years after discharge, mainly due to the smaller fission products content arising from a lower discharge burnup. However, the radiotoxicity at a few hundred years is higher than those of the other reactor-based transmutation cases because of the higher amount of Am-241 produced in the RMWR case.

By recycling both the plutonium and uranium, the RMWR concept has excellent features in terms of the uranium utilization and waste reduction. However, the tradeoff for this benefit is an increase in the amount of heavy metal to be separated and reprocessed per unit electricity production.

3D. Criticality Benchmarks

Session Organizers: Russell D. Mosteller (LANL), Hironobu Unesaki (Kyoto University).
Session Chairs: Richard D. McKnight (ANL), Hironobu Unesaki (Kyoto University).

TransLAT Lattice Physics Code Benchmark to B&W Gadolinia Criticals

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The TransLAT three-dimensional lattice physics code has been benchmarked to 19 critical experiments performed at the B&W Research Center in 1984. The experimental core configuration consisted of a 3x3 array of PWR 15x15 fuel assemblies in the center of the core surrounded by a driver region. The cores were constructed with 4,961 pin locations containing combinations of UO_2 fuel pins, gadolinia fuel pins, Ag-In-Cd and B_4C absorber pins, water holes and void pins. Criticality was achieved by varying the moderator boron concentration.

The transport calculations in TransLAT were performed in 2-D planar geometry using the Method of Characteristics solution technique with a 36-group ENDF/B-VI.5 library. The k-effective was calculated for each of the nineteen cores, and comparisons of the calculated and measured fission rates for the center assembly were performed for four of the cores. The average k-effective is 1.00013 with a standard deviation of 0.00056 for the 19 core configurations. The maximum difference between the measured and calculated fission rates for the four core configurations with measurements is 0.025. The results demonstrate that TransLAT is accurately calculating the reactivity and fission rate distributions for these gadolinia criticals.

ENDF/B-V and ENDF/B-VI Calculations for the LWBR SB Core Benchmarks with MCNP5™

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Recently, detailed benchmark specifications have been issued for the Light Water Breeder Reactor (LWBR) Seed and Blanket (SB) critical experiments. These benchmarks are of particular interest because they include critical lattices of ^{233}U and highly enriched uranium fuel pins completely immersed in water. All of these benchmarks have thermal spectra.

MCNP5 calculations have been performed for the eight benchmarks in the set, using ENDF/B-V and ENDF/B-VI nuclear data libraries. ENDF/B-V produces good agreement with the benchmark values for k_{eff}. ENDF/B-VI produces reasonable agreement with the benchmark values but not as good as ENDF/B-V. However, the ENDF/B-VI results are less sensitive to the H/U ratio than the ENDF/B-V results are.

Both ENDF/B-V and ENDF/B-VI substantially underpredict k_{eff} for the three cases that have $^{233}UO_2$-ThO_2 blankets. The thorium in the blanket rods contains a small amount of gadolinium, and the reactivity worth of the gadolinium varies from approximately -0.002 Δk to approximately -0.004 Δk for the three cases. Given the ambiguity in the gadolinium content and its reactivity impact, it is recommended that the uncertainty associated with the benchmark value of k_{eff} for these three cases be increased.

3D. Criticality Benchmarks

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Session Chairs: Richard D. McKnight (ANL), Hironobu Unesaki (Kyoto University).

Benchmark Comparisons of Deterministic and Monte Carlo Codes for a PWR Heterogeneous Assembly Design

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Numerical benchmark calculations for a heterogeneous CORAIL assembly have been performed using deterministic transport codes (WIMS8 and APOLLO2) and Monte Carlo codes (MCNP4C and TRIPOLI4). For this benchmark, the eigenvalues, normalized pin power distributions, and the fuel inventory calculated using several nuclear data files such as ENDF/B-V, -VI, and JEF2.2, were compared.

Regarding the assembly eigenvalue, the differences between the APOLLO2, TRIPOLI4 and MCNP4C results are satisfactory when cross sections based on the same nuclear data file are used (deviation less than 200 pcm (Δk)). A significantly higher difference in the k_∞ value (up to 500 pcm) is observed between the Monte Carlo cases using cross sections based on different evaluated nuclear data files (particularly JEF2.2 versus ENDF/B-V or –VI). Although the heterogeneous fuel pin configuration results in a sharp flux gradient in the assembly, there is good agreement between the normalized power distributions calculated by the codes.

A good agreement in the fuel inventory calculated by the two deterministic codes is also observed.

Analysis of the Experimental Program MISTRAL using CASMO-4

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An experimental program “MISTRAL” was undertaken by NUPEC, CEA and their associated industrial partners to investigate the main neutronic characteristics of high moderation ALWR loaded with 100% MOX fuel. The MISTRAL program comprises four core configurations of one homogeneous UO_2 core, two homogeneous and one heterogeneous MOX cores. The analyses of the MISTRAL experiment were performed to evaluate the capability of CASMO-4 code by NUPEC. The critical k_{eff} was overestimated for MOX cores. For the fission rate distribution the results agreed well with the experimental data. For absorber worth, boron efficiency coefficient, temperature coefficient, water-hole reactivity and 2D void reactivity the results were predicted within about twice the experimental uncertainty. CASMO-4 had the same accuracy in all configurations except for the critical k_{eff}.

Solution of the C5G7 3-D Extension Benchmark by the S_N Code TORT

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In the present paper, we report on our solution of the 3-D extension of the C5G7 MOX fuel assembly benchmark obtained by the 3-D multigroup discrete ordinates transport code TORT. Main feature of this benchmark is a (quarter core symmetric) sixteen assembly core with an arrangement of UO_2 and MOX fuel assemblies surrounded by a water reflector. Each assembly consists of a 17×17 lattice of square pin cells with circular fuel rod and surrounding moderator. Since no homogenisation technique should be applied, in our calculation the fuel rod is nodalised by a cartesian mesh. For the three specified control rod configurations, both the effective multiplication factors and the pin-wise fission rates obtained in our calculations are in good agreement with the given Monte-Carlo reference solution. Different sets of spatial meshes are used to study the sensitivity of the results on the pin cell nodalisation. We find that there is a stronger dependence on the spatial than on the angular variable, represented by the quadrature order. In addition, we studied the

effect of the control rods on the axial scalar fluxes and fission rates and found that, depending on their insertion depths, the cosine-like flux shape of the uncontrolled core is significantly modified by the control rods.

TORT Solutions to the Three-Dimensional MOX Neutron Transport Benchmark Problem, 3-D Extension C5G7 MOX

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This paper details the results for the OECD/NEA C5G7MOX benchmark obtained using the DOORS code TORT. This benchmark is intended to test modern radiation transport codes' abilities to handle three-dimensional problems with spatial heterogeneity. The problem consists of a four-assembly UO_2 and MOX fuel reflected reactor core in which control rods are inserted to three distinct levels, *Unrodded, Rodded A*, and *Rodded B*. A Monte Carlo reference solution was provided with the benchmark. Effective multiplication factor results varied from 0.06% error for the *Unrodded* case to 0.17% and 0.67% for the *Rodded A* and *Rodded B* cases, respectively. The pin power results followed a similar trend. Varying the meshing and angular quadrature refinement did change the results, but not sufficiently to justify the additional computation time.

Creation of a Simplified Benchmark Model for the Neptunium Sphere Experiment

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Although neptunium is produced in significant amounts by nuclear power reactors, its critical mass is not well known. In addition, sizeable uncertainties exist for its cross sections. As an important step toward resolution of these issues, a critical experiment was conducted in 2002 at the Los Alamos Critical Experiments Facility. In the experiment, a 6-kg sphere of ^{237}Np was surrounded by nested hemispherical shells of highly enriched uranium.
The shells were required in order to reach a critical condition.

Subsequently, a detailed model of the experiment was developed. This model faithfully reproduces the components of the experiment, but it is geometrically complex. Furthermore, the isotopics analysis upon which that model is based omits nearly 1% of the mass of the sphere.

A simplified benchmark model has been constructed that retains all of the neutronically important aspects of the detailed model and substantially reduces the computer resources required for the calculation. The reactivity impact of each of the simplifications is quantified, including the effect of the missing mass. A complete set of specifications for the benchmark is included in the full paper.

Both the detailed and simplified benchmark models underpredict k_{eff} by more than 1% Δk. This discrepancy supports the suspicion that better cross sections are needed for ^{237}Np.

Monte-Carlo Techniques to Simulate Pebble Dislocations in a PB-MR during Depletion

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A Pebble Bed Modular Reactor (PB-MR) is made of a bulk of small pebbles having, each, an independent history during its downward motion within the core and, as consequence, achieving, at each instant of the reactor life, a different burn-up. The pebble burn-up distribution within the PB-MR has an immediate, measurable impact on reactivity, core stability and cycle performance. Thus, it is worth accounting for it in a simplified, but effective way when performing process studies of such cores. The methodology described here is aimed at simulating the depletion of the pebbles during their cyclical motion in the core by using a Discrete Time kinetic Monte Carlo. For this purpose the totally new code DISLOC was developed and the preliminary simulations results, including burn up evolution, on the dynamic of the pebbles movements were presented.

Design of a Very High Temperature Pebble-Bed Reactor Using Genetic Algorithms

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Efficient electricity and hydrogen production distinguish the Very High Temperature Reactor as the leading Generation IV advanced concept. This graphite-moderated, helium-cooled reactor achieves a requisite high outlet temperature while retaining the passive safety and proliferation resistance required of Generation IV designs. Furthermore, a recirculating pebble-bed VHTR can operate with minimal excess reactivity to yield improved fuel economy and superior resistance to ingress events. Using the PEBBED code developed at the INEEL in conjunction with a Genetic Algorithm for core optimization, conceptual designs of 300 megawatt and 600 megawatt (thermal) Very High Temperature Pebble-Bed Reactors have been developed. The fuel requirements of these compare favorably to the South African PBMR. Passive safety is confirmed with the MELCOR accident analysis code.

Feasibility of Using Burnable Poisons for Reduction of Coolant Void Reactivity in LMR for TRU Transmutation

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A new design concept to reduce coolant void reactivity in sodium cooled core for transuranic element transmutation is proposed. In the new option, some amount of fertile material is removed for reduction of sodium void reactivity. Simultaneously, a burnable absorber material is loaded in replacement of fertile material to compensate for reactivity drop during the fuel depletion. Considering two methods of burnable absorber loading such as the homogeneous and the heterogeneous, the feasibility of the new option using burnable poison are discussed in terms of sodium void reactivity, burnup reactivity, and Doppler effect. In the results, it is found that the homogeneous loading cannot reduce the sodium void reactivity but makes the reactivity more positive. On the other hand, the heterogeneous loading can reduce the sodium void reactivity successfully. It is also noticed that the increment in burnup reactivity swing is negligible when the burnable poison is heterogeneously loaded in the central region of the core. The obtained results lead to the conclusions that if the burnable poison material is loaded appropriately, the sodium void reactivity can be reduced without any significant penalty of increase in burnup reactivity swing.

Plutonium Disposition in the PBMR-400 High-Temperature Gas-Cooled Reactor

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Versatility in application is considered to be one of the major attributes of a typical Generation IV type reactor. This characteristic is exploited in a 400 MW_{th} layout of the Pebble Bed Modular Reactor design currently under development in South Africa. The versatility offered by the multi-pass, on-line pebble fuelling concept is demonstrated in a design variation aimed at the dispositioning of reactor grade plutonium or weapons grade plutonium.

For the purposes of comparative investigation two cases have been studied, viz. the so-called Reactor-Pu case with a Plutonium-Thorium/Uranium fuel cycle and the Weapons-Pu case with a Plutonium/Thorium fuel cycle. In both instances a dual-pebble type layout is proposed where one type of pebble is charged with the Pu to be dispositioned, while the driver pebbles contain the Th and highly enriched U as a mixed oxide. Both pebble types are produced in the usual configuration consisting of a sphere with 60 mm diameter, containing the fuel zone of 50 mm diameter that in turn contains the so-called TRISO-coated particles. For the Reactor-Pu case a 5-pass cycle is employed whereas a 4- and 6-pass cycle is proposed for the Weapons-Pu case. This is practically viable due to the different physical characteristics of the various pebble types that can be measured or mechanically observed.

Calculated results of the destruction values achieved and coefficients of reactivity are discussed.

Definition of a Calculation Scheme with MONTEBURNS for Decay Heat Calculations of High Temperature Reactor Fuel

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To carry out design studies of prismatic High Temperature Reactor (HTR) cores, it is necessary to calculate the residual heat and the activity of HTR spent fuel. The computer code used by FRAMATOME ANP to achieve such studies is MONTEBURNS, a code linking ORIGEN2 and MCNP, which enables to model all the heterogeneities of the fuel.

The calculations are performed for an infinite lattice of GT-MHR prismatic fuel assemblies.

The first part of the study consists in optimizing the input parameters of the calculations (the number of depleting materials, the number of burn steps, the convergence of MCNP calculations and the number of isotopes transferred between MCNP and ORIGEN2). Then the impact of the approximations concerning the MCNP and ORIGEN2 cross sections libraries, and of the geometry of the model, is quantified. The analysis of the calculation results shows that MONTEBURNS enables to obtain accurate results within a reasonable calculation time.

Concept of a Gas Cooled Fast Reactor

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Currently in a number of the countries the development of reactor designs for the following generation is being carried out.

Among reactors of a new generation the direction of high-temperature gas cooled reactors (both thermal, and fast – GFR), in particular, is recognized.

The purpose of the given work is research the opportunities of HTGRs with a hard neutron spectrum to increase the duration of reactor campaign due efficient neutrons usage at preservation of specific core power density, characteristic for HTGRs with a thermal neutron spectrum. The given purpose can be achieved at use fuel blocks with dense packing of coated particles in the fuel block volume.

To solve the presented problem the variant of so-called "return" fuel assembly is proposed where the coated fuel particles in fuel block matrix material occupy all volume of the block except for standard channels under the coolant. In this case the volume fractions of materials in prismatic type assembly will take approximately 19/31/50 %% accordingly for the coolant, fuel kernels and matrix including the coatings. As results of unit-cell calculations show the small rate of reactivity falling presumes to compensate the reactivity margin by control rods without burnable poison using. The core campaign is comparable with service life of power plant and can consists of 40 years.

MINERVE Reactor Characterization in Support of the OSMOSE Program: Safety Parameters

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The experimental reactor MINERVE is devoted to neutronic studies of lattices of different reactor types. MINERVE is a pool type reactor operating at a maximum power of 100 W. The core, submerged under 3 meters of water, is used as a driver zone for the different experiments located in a central square cavity with a size of about 70 cm × 70 cm. The coupled lattices in this cavity are built in such a way that they can reproduce the neutronic spectra of a fast reactor, a light water reactor, a RSM reactor and a heavy water reactor. It provides a large experimental basis for the improvement of the cross section databases. An oscillation technique is used to determine the reactivity worth of samples containing the material of interest: actinides, absorbers, poisons, spent fuel, structural materials, etc.

An ambitious program between the Commissariat à l'Energie Atomique (CEA) and the U.S. Department of Energy (DOE) has been launched with the aim of measuring the integral absorption rate parameters of actinide isotopes in the MINERVE experimental facility located at the CEA Cadarache Research Center. The OSMOSE Program (OScillation in Minerve of isOtopes in "Eupraxic" Spectra) includes a complete analytical program associated with the experimental measurement program and aims at understanding and resolving potential discrepancies between calculated and measured values for the studied actinides.

Before beginning the experimental measurements, it is necessary to prove that it is possible to accurately predict the safety parameters for the reactor operation and to effectively control the reactor during experimental measurements. In this context, a significant effort is underway to model the reactor, to calculate relevant safety parameters, and to conduct measurements using the MINERVE reactor to obtain values for the safety parameters.

This paper describes efforts and results of calculations and measurements of the safety parameters. Specifically, the multiplication factor, the reactivity worth of the control rods, the critical position of the rods, the reactivity excess of the core, the axial profile of the reaction rates in the oscillation position, and the radial power profile inside the experimental region are being studied. Results of calculations are also compared to measurements obtained for the parameters.

Thermo Mechanical Calculations for Integrated High Fidelity Reactor Core Simulations

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The Numerical Nuclear Reactor is a collaborative US-ROK I-NERI project to develop a comprehensive high fidelity reactor core modeling capability for detailed analysis of current and advanced reactor design. One of the objectives of the US-ROK collaborative I-NERI project known as the "Numerical Reactor" is the application of thermo-mechanical techniques for structural calculations as part of integrated whole-core simulations. In the thermo-mechanics area, activities to date are focused on assessment of the thermo-mechanical response of fuel assemblies, development of an efficient computational methodology to simulate that response, and establishing an interface for coupling of this methodology with the CFD module. The mechanical response or bowing deflection of core assemblies due to thermal gradients, swelling and irradiation creep are being formulated as a function of location in the core and the assembly support structures for a typical PWR core. It is anticipated that these mechanical responses will be significant during startup, long term operation, and various transients, and examination of the effects of such geometric changes on reactor operations will be important.

Monte Carlo Midway Forward-Adjoint Coupling with Legendre Polynomials for Borehole Logging Applications

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The Monte Carlo Midway method utilizes the surface integral form of the reciprocity theorem in order to increase computational efficiency. The response of a detector is calculated by integrating the product of the adjoint function and the flux regarding all phase space variables, whereas both functions are stochastically sampled. Formerly, coupling of the two functions has been done by estimating the flux and the adjoint functions in small phase-space segments. In this paper, we investigated the possibility to use Legendre polynomial expansion on a stochastic basis for the surface integral. A formula for the variance of the expansion coefficients is derived. Using a simple mathematical example the systematic error as a function of the number of segments or the Legendre expansion order is investigated. The new algorithm is tested with a borehole logging application, estimating the time-dependent response of a photon detector due to a 14 MeV neutron source. It resulted in a higher accuracy of the forward-adjoint coupling, but the efficiency of the calculation decreased.

Visualization of Space-Dependency of Reponses of Monte Carlo Calculations Using Legendre Polynomials

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Visualization of space dependency of quantities of interest is not typically a task to set for Monte Carlo calculations. Often it is, though, useful and desired. Most commonly, it is done by segmenting the phase space into meshes on a surface, and obtaining estimates in each subregion.

In this paper we investigate the possibility of using an orthogonal function basis for visual description of the quantities of interest and the determination of the coefficients by Monte Carlo. The basic formulas for determining the expansion coefficients and their variance from a Monte Carlo calculation are given.

To test the flux expansion method, first a simple geometry problem is considered to represent the radial flux in a plane. Next, a more complicated geometry is chosen and a reconstruction of the neutron flux in a plane using Legendre polynomials is compared with the classical method of subdivision in segments and scoring the average flux over each segment. The Legendre expansion method shows more detail in the flux reconstruction. Using Laguerre polynomials for the flux expansion did not work out satisfactory.

An Influence of Core Physics Peculiarities upon the Thermal Hydraulics Performance in Cascade Subcritical Molten Salt Reactor

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This paper presents the results of investigation of the influence of core physics peculiarities upon thermal hydraulics and workability of the main structures in Cascade Subcritical Molten Salt Reactor. Different variants of the target have been considered. Revealed it has been that the optimal variant of the target design (from point-of-view neutron efficiency + coolability) corresponds to the tight tungsten rod bundle packed in hexagonal lattice cooled by the same

molten salt as being circulated in the central core zone. The limit of the power volumetric density in the tungsten rods has been defined from the thermal physics and thermal mechanics calculational estimations. The results are demonstrated of research of the central core zone thermal physics for a few variants differed by the combinations of the coolant, structures and fuel in the central zone. Different variants of circulation (forced and natural) of the coolant in the central zone have been considered from view-point of workability of the central zone structures under condition of the great power rate. The questions of workability of the Intermediate Heat Exchangers of the central and peripheral core zones for different variants of reactor design are also discussed.

AGENT Code: Open-Architecture Analysis and Configuration of Research Reactors – *Graphic Tools*

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Two papers presented in this Conference focuses on our recent development of the advanced computational environment for the analysis and configuration of University Research and Training Reactors (URTRs). This computational environment is intending to assist researchers and educators with tools for an open-architecture neutronic analysis and configuration of the URTRs. Such computational environment does not currently exist. The method of characteristics based computer code, AGENT will revolutionize the way in which URTR research is planned, deployed and analyzed through the so called "*virtual reactor environment*" that is planed to be a fully open modular code system for access and use by the entire URTR and associated reactor communities. Main applications target the basis for a variety of experimental and component design activities, and a useful aid for a number of out-reach activities at all educational levels. The platform is supposed to be accessible by K through 12 schools via the internet, so that realistic URTR reactor performance cases can be run and viewed on-line from their classrooms. As we expect this may provide a major new resource to the nation for the widespread development of educational and research activities. One of major issues related to a successful usage and application of the complex virtual computer simulation tools is a graphic support. This paper provides an overview of the graphic tools that were designed to support the AGENT code programming architecture.

DELPHI: A New Subcritical Assembly at Delft University of Technology

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This paper describes the new subcritical assembly called DELPHI at Delft University of Technology. It consists of two vessels one upon the other, and contains 168 fuel pins made of 3.8 % enriched UO_2 fuel. The upper acrylic vessel is used to store the fuel pins. With a special handling tool the fuel pins can be loaded into the water-filled steel vessel. The source is stored in a shielding box below the steel vessel and can be inserted pneumatically up to 2 cm below the fuel zone of the pins.

DELPHI became operational at the beginning of 2004 and will be used for training of students and reactor operators from the Netherlands and abroad and for basic research on reactivity determination methods.

The first experiment carried out was the approach to critical. For this experiment, the fuel was loaded in batches of 8 pins, and after each loading the detector count rates were measured at various positions with and without the source. Because the source is located below the fuel zone, the spatial flux distribution shifts strongly during loading. For detector positions too close to the source, this gives a strong underestimation for the k_{eff} derived from the multiplication factor.

The Studies of RBMK-1500 Reactor Core Behavior during Abnormal Operation Transients

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This paper describes series of RBMK-1500 reactor transient investigations, performed with CORETRAN code. Aspects of the reactor core neutronic and thermal hydraulic behavior during postulated core transients were analyzed. In particular, RBMK-1500 transients leading to changes in reactor power and core reactivity were considered. Three reactivity-initiated accident cases were addressed: a) spontaneous control rod bank withdrawal in the central part of the core; b) spontaneous control rod bank withdrawal in core periphery and; c) release of one Shortened Absorber Rod from the reactor core. Reactor nominal power operation was used as the reference core state. Analysis was performed using the data, obtained from the actual plant database recorded for Ignalina Nuclear Power Plant Unit 2 on 27th January 2001. The CORETRAN calculations were benchmarked against STEPAN code results.

Onboard Radiation Shielding Estimates for Interplanetary Manned Missions

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One of the main deterrents to manned interplanetary missions, like the one to Mars, is the time a crew would spend in space traveling to and returning from the planet. Several new propulsion designs have been theorized to shorten this time, thereby increasing the feasibility of a mission. The idea of using a nuclear powered propulsion system seems to be recently the most promoted option for space travel. Radiation (both external and on-board) is one of the major hazards for astronauts in space and has emerged as a most critical issue to be resolved for long-term flights. The main focus of space related shielding design is to protect operating systems, personnel and key structural components from outer space and onboard radiation. This paper summarizes the feasibility of a lightweight neutron radiation shield design for a nuclear powered, manned space vehicle. The Monte Carlo code MCNP5 is used to determine radiation transport characteristics of the different materials and find the optimized shield configuration. A phantom torso encased in air is used to determine a dose rate for a crew member on the ship. Calculation results indicate that onboard shield against neutron radiation coming from nuclear engine can be achieved with very little addition of weight to the space vehicle. The selection of materials and neutron transport analysis as presented in this paper are useful starting data to design shield against neutrons generated when high-energy particles from outer space interact with matter on the space.

Reduction of Cross-Section-Induced Errors of the BN-600 Hybrid Core Nuclear Parameters by Using BFS-62 Critical Experiment Data

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The present paper provides evaluation results of predicted uncertainty on nuclear parameters on the BN-600 hybrid core, a feasible option for Russian surplus weapons plutonium disposition. In order to support the option, Japan Nuclear Cycle Development Institute (JNC) has performed a series of measurements using BFS-2 facility of Russian Institute of Physics & Power Engineering (IPPE) under a collaborative research, by assembling several core configurations that simulate nuclear characteristics of the possible BN-600 core transition. The experiment and analysis results were already presented in PHYSOR2002.

Successively, by reflecting JNC's experimental analysis results of both the BFS-2 and other fast reactor core critical assemblies, uncertainties of key nuclear parameters on the BN-600 hybrid core were predicted by the nuclear group constant adjustment method, as well as the bias correction method. Covariance of JENDL-3.2-based nuclear group constant set, analysis error, and experimental error are considered to predict the uncertainties.

It is concluded that the nuclear constant adjustment method can largely reduce the uncertainty for all the discussed nuclear parameters. In addition, significance of BFS-2 mockup data, effects of cross-section covariance changes on the uncertainties, and comparison of predicted uncertainties among methods are mainly discussed.

Uncertainty Analysis Results on BN-600 Hybrid Core Nuclear Physics Characteristics

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The uncertainty evaluation results for the nuclear physics parameters of the BN-600 advanced hybrid core are presented. For the evaluation, the analysis results of BFS-62 series of fast reactor benchmark experiments, mock-ups for possible transition of future BN-600 cores, together with other benchmark experiments are used. One of the main items of the BN-600 core design is the replacement of uranium blanket by the steel reflector. But, considerable inconsistency between the calculated and measured values (20% and more) was found for the fission reaction rate distribution outside the core, in the steel reflector region of the BFS-62 mock-up cores. That, however, proves the results obtained before in analyzing CIRANO ZONA2B experiments performed in MASURCA. The part of the uncertainty associated with the nuclear data is estimated by the adjustment method using different covariance evaluations: ABBN-78, ENDF/B-V, ENDF-B-VI, JEF-2.2 and JENDL-3.2. The most significant differences are found in JENDL-3.2 case which gives the most optimistic results. The conclusion is made about the prediction uncertainty of the main BN-600 hybrid core nuclear physics parameters. It is underlined that because of the found inconsistency between calculation results and measurements in the steel reflector region, which can not be explained by experimental error or treatment approach, this problem has a fundamental consequence as well as it is important for the shielding, so it has to be studied more.

Analysis of SNEAK-7A & 7B Critical Benchmarks using 3-D Deterministic Transport and Sensitivity Analysis Methods

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As the detailed information on critical experiments performed in the SNEAK facility became available in the framework of the International Reactor Physics Benchmark Experiments (IRPhE) project, SNEAK-7A & 7B critical benchmarks are re-evaluated. The realistic modeling of these assemblies was done using the DANTSYS code capability for X-Y-Z geometry.

Predicted core eigenvalues, spectral indices, and central material worths with effective cross sections based on JEF-2.2 nuclear data file, have been compared with measured values. As a result, the capability of modern 3-D transport method featuring very small corrections to the raw calculated value is appreciated and the quality of basic library for the interested assemblies is assessed.

In addition, the core eigenvalue sensitivity analysis to the basic cross section of each important nuclide is performed. By utilizing the deviation of calculated spectrum indices from the measurement, together with the sensitivity coefficient, it is shown how the sensitivity analysis can be usefully adopted to figure out a specific detail of basic cross section for improvement and investigate the quality of measured integral parameters.

Derivation of the Space and Energy Dependent Formula for the Third Order Neutron Correlation Technique

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We have studied a measurement technique of subcriticality by using the neutron correlation methods. Among various techniques, we have paid attention to the third order neutron correlation technique, which utilizes the third order fluctuations of neutron counts. By using this technique, we can obtain the absolute value of subcriticality without prior knowledge of the prompt neutron decay constant at a reference state. To apply the third order neutron correlation technique to actual experiments, we must consider the effects of the spatial and neutron energy distributions in this technique. For this purpose, we derived the generalized theoretical formulas of the third order neutron correlation technique that took account of the spatial and neutron energy effects.

Cross-Section Analysis for TRADE Fuel

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The TRIGA core includes bounded hydrogen in Zirconium hydride in its fuel meat allowing for fast reactivity transients. The inherent safety mechanism is based on the immediate increase of neutron up-scattering by the hydrogen as a result of a fuel temperature increase. The temperature dependent resonance absorption is the second safety feature. The special fuel type together with the introduction of an external source within it for the TRADE project necessitates an accurate evaluation of the bounded hydrogen cross section generation technique as well as of the resonance treatment. By comparing deterministic tools and Monte Carlo solution methods the generated bounded isotopes cross sections are analysed. Further, the importance of the Doppler and the thermal up-scattering effects are quantified and the sensitivities to the solution method are discussed. From the current study it is concluded that the cross section generation technique for deterministic codes agree in general with criticality Monte Carlo solution and the differences in the criticality values is below 1%. A more detailed analysis using several ENDF data libraries shows however that in the MCNP data processing there are some inaccuracies. Other sources for discrepancies between the deterministic and stochastic methods are the heterogeneity treatment (geometrical cross section) and the resonance absorption and its effective group averaged macroscopic cross section derivation.

On-line Determination of the Prompt Fraction of In-Core Neutron Detectors in CANDU Reactors

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This paper describes a new method for determining the prompt fraction of in-core neutron detectors in CANDU reactors. The method is based on noise analysis, and thus does not require any perturbation of the reactor operation. This method is therefore very well suited to on-line monitoring, and could detect early detector degradation as the detectors age. The prompt fraction estimation of the in-core neutron detectors is of prime importance in CANDU reactors, since these detectors are used by the reactor shutdown systems, and should consequently respond very quickly to any flux change.

This new method is based on the fluctuations of the light water levels in the water compartments at a frequency of roughly 0.25 Hz. These fluctuations are due to the control cycle of the regulating system of the reactor, and are equivalent to a spatially-distributed noise source of variable strength. The induced neutron noise can be monitored by the in-core neutron detectors, but only the prompt component of these detectors is able to follow these fluctuations at

0.25 Hz. Comparing the measured detector signals to the neutron noise estimated by core calculations allow determining the prompt fraction of the detectors.

Dynamics of a Reduced Order Natural Circulation BWR Model

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Stability and bifurcation analyses of a new natural circulation BWR model that allows local pressure dependence of water saturation enthalpy and includes fundamental and first azimuthal modes for neutronics are presented. Stability boundaries (SBs) and 7.5% oscillation curves (representing natures of bifurcation along SBs) are plotted in inlet subcooling—external reactivity parameter space. Stability and bifurcation analyses show that both in-phase and out-of-phase oscillations as well as supercritical and subcritical bifurcations can occur along the SBs. Results of numerical simulations agree with those of stability and bifurcation analyses. Effects of several design and operating parameters on the stability of a natural circulation BWR are also reported.

Solution of the 1D Kinetic Diffusion Equations Using a Reduced Nodal Cubic Scheme

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In this work a nodal method, based in a cubic polynomial approximation, has been developed to solve numerically the time-dependent 1D multi-group neutron diffusion equations. The cubic approximation is built using as interpolation parameters the zero-th and first Legendre moments of the dependent variables (flux and precursors concentrations) and its point values at the left and right edges of each cell of the discretization. Following the Galerkin finite element procedure a set of ordinary differential equations is obtained which is finally discretized applying the θ scheme. The elements of the stiffness and mass matrices are evaluated applying two quadrature rules to decouple left and right unknowns of each element of the spatial discretization allowing to solve the point values of the flux in terms of the zero-th and first Legendre moments of the flux of two contiguous cells of the discretization. This action permits to get a reduced set of equations having only the zero-th and first order Legendre moments of the flux per cell as unknowns. For each time step these Legendre moments of the flux are obtained. Then the corresponding delayed precursors are computed before a new time step be initiated. Numerical results for a well known benchmark problem are given.

Implementation of a Full P_1 Method in the Diffusion Code DONJON/NDF

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Rigorous diffusion theory is obtained by the spherical harmonics expansion of the angular variable in the Boltzman transport equation, truncated to first order terms. This is known as the P_1 approximation. Diffusion coefficients do not arise naturally from this approach, and are not necessary at all. The relationships between fluxes and currents appear through terms involving anisotropic scattering cross-sections. This method gives a more accurate representation of the Boltzman equation than standard diffusion theory. The P_1 method enables slightly more accurate calculations at heterogeneous interfaces. The method was programmed as a part of the DONJON/NDF package for both the static and dynamic cases in multi-group mesh centered finite differences. The coupling coefficients that are diagonal matrices with diffusion coefficients become full matrices because of the anisotropic scattering cross-sections. Several comparisons between standard diffusion theory and the P_1 approximation are presented, including some simple transients of a typical CANDU-6 reactor model.

The Even-Parity Simplified S_N Equations Applied to a MOX Fuel Assembly Benchmark Problem on Distributed Memory Environments

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Criticality calculations for nuclear reactors design are usually performed using the diffusion equation with homogenized cross-sections. The diffusion approximation yields accurate results for problems where the particles flux present a weak angular dependency. However, reactor cores loaded with Mixed Oxide (MOX) fuel assemblies are characterized by strong absorption in the thermal range due to Pu-239 and the interface between fuel and moderator introduces strong angular dependencies in the flux. Because of these aspects, the diffusion equation presents low accuracy. To address this problem, we have investigated the accuracy of the Even-Parity Simplified S_N (EP-SS_N) equations. The EP-SS_N equations allow for higher order angular dependencies, therefore it is expected that the methodology will be more accurate compared to the diffusion model for problems characterized by strong transport effects such as MOX loaded reactor cores.

This paper discusses a new general 3-D parallel code, $PENSS_N$ (Parallel Environment Neutral-particle Simplified S_N), based on the multigroup even-parity form of the SS_N equations. Further, it will present the results obtained for a MOX benchmark problem proposed by OECD/NEA and estimate the accuracy of the EP-SS_N equations compared with the reference Monte Carlo predictions.

On the Influence of Differences between Various Group Microconstant Libraries and between Different Transport Options on Calculation Results for Cells and Subassemblies of VVER-1000 Reactor

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We have carried out a series of calculations for VVER cells with different nuclide compositions and for two subassemblies composed of these cells. Various transport options for WIMS-D4 [1] and WIMS-7B [2] codes were applied with the use of group microconstant libraries contained in these codes. Then we carried out similar calculations using our new code SVL where the algorithm for transport computations was based on the Surface Pseudo-Source Method (SPSM) for cells and Surface Harmonic Method (SHM) for subassemblies [3]. The microconstant library is based on UNKDL files (WIMSD4) and on JEF 2.1 files for some nuclides. The library like with one of those built into the WIMS-7B code (86.69) where the first number shows the year this library has been initially used and the second number indicates the number of groups.

Results obtained allow one to draw some important conclusions regarding the influence that libraries and transport options might have on calculations of VVER-cells and subassemblies' characteristics.

The following conclusions may be drawn from our analysis.

1. The library (69.86) gives quite satisfactory results.
2. The difference between these results and those obtained using the library (97.172) is less than 0.2% for multiplication factors.

At present the numerous calculations of different tasks have been performed (exemplarily 250 cells, fuel and critical assemblies, with variants, taking into account changes temperatures, boron concentrations in moderator more than 1 000). Only small part of these results might be presented in this report. After comparison of our results with those of other authors and with experiments one can draw the following conclusion. In spite of the calculation time by the SVL code is less on orders, sometimes on many orders in comparison with calculation time of another codes, the SVL code do not yield to the best benchmark codes on accuracy of calculation.

Sipping Tests on a Failed Irradiated MTR Fuel Elements

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This work describes sipping tests performed on Material Testing Reactor (MTR) fuel elements of the IEA-R1 research reactor, in order to find out which one failed in the core during a routine operation. Radioactive iodine isotopes ^{131}I and ^{133}I, employed as failure monitors, were detected in samples corresponding to the failed fuel element. The specific activity of each sample, as well as the average leaking rate, were measured for ^{137}Cs. The nuclear fuels U_3O_8 – Al dispersion and U – Al alloy were compared concerning their measured average leaking rates of ^{137}Cs.

Capturing the Effects of Unlike Neighbors in Single-Assembly Calculations

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We present recent improvements in assembly-level calculations for reactor analysis, including modifications that support core-level analysis by *quasi-diffusion*. Our main focus is on accurately approximating the effects that neighboring assemblies have on the few-group cross sections, assembly discontinuity factors, form factors, and other transport parameters of a given assembly. We show that we can do this by using albedo boundary conditions that are estimated with low computational cost. We also present an efficient way to tabulate these effects to permit accurate interpolation by the core-level algorithm. We describe our algorithms and present results from several difficult test problems containing MOX and UO_2 assemblies. Our methodology significantly reduces the largest errors made by present-day methodology. For example, in our test problems it reduces the maximum pin-power error by approximately a factor of 5.

New Computational Methodology for Large 3D Neutron Transport Problems

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We present a new computational methodology, based on 3D characteristics method, dedicated to solve very large 3D problems (a part or a whole core) without spatial homogenization. In order to eliminate the input/output problems occurring when solving these large problems, we set up a new computing scheme that requires more CPU resources than the usual one, based on sweeps over large tracking files. The huge capacity of storage needed in some problems and the related I/O queries needed by the characteristics solver are replaced by on-the-fly recalculation of tracks at each iteration step. Using this technique, large 3D problems are no longer I/O-bound, and distributed CPU resources can be efficiently used.

Spent Nuclear Fuel Analyses Based on In-Core Fuel Management Calculations

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A method for performing spent nuclear fuel (SNF) analyses using in-core fuel management (ICFM) calculations has been developed and validated as described in the full paper. Parameters such as decay heat, neutron source rates and photon spectra of discharged LWR fuel assemblies are calculated. This approach offers highly automated and accurate SNF analyses and eliminates the need for special fuel irradiation calculations to obtain the radiation source terms.

The isotopic contents of the fuel at the time of discharge from the reactor are obtained from standard ICFM core tracking calculations. The spent fuel source terms, at any given time after discharge, are then calculated by solving the isotopic decay chains and using the isotope summation method. The ENDF/B-VI and JEF-2.2 data libraries of the lattice codes CASMO-4 and HELIOS include a sufficiently large number of actinides and fission products for successful application of this method. Examples of validation showing this are described in the paper.

The operating history of a given fuel assembly is obtained from the ICFM 3-D simulation, e.g. using SIMULATE-3 core follow calculations. The isotopic concentrations to be used in the SNF calculation are thus evaluated on a nodal level, based on the final burnup, neutron spectrum history and power density history of each axial node of a given fuel assembly. The required library data were obtained from the ENDF/B-VI Decay Data File, supplemented with other data.

ALAADIN/FLS – A BWR Fast Lattice Design Simulation Tool

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This paper describes a BWR fast lattice design simulation tool - ALAADIN/FLS which will be used in a BWR assembly lattice optimization (BALO) system. ALAADIN is a graphical user interface. FLS, a fast lattice simulator, provides a quick solution of three nuclear characteristic curves of lattice at three voids (0%, 40% and 80%): infinitive multiplication factors (Kinf) vs. burn up, maximum pin power vs. burn up, F-eff vs. burn up. FLS employs the following algorithms: massive non-linear two dimensional interpolations with a perturbation library, superposition and pin power re-normalization. For each single fuel pin, the non linear interference between variations of enrichment and Gd weight is taken into account in the algorithms. Among pins, the enrichment perturbations show that the linear superposition is a good approximation. But among Gd pins, the linear superposition is good only for sparse and scattered Gd pins, not for Gd clusters. A correction matrix must be added to the perturbation library to accommodate Gd clusters. For normal designs with no Gd clusters, FLS has shown very good results over the entire design range from 0 GWD/MT to 70 GWD/MT for enrichments from 0.71% to 5% and Gd weights from 0% to 8%. Compared with

CASMO4, the time to obtain results is measured in seconds instead of hours, with Kinf accuracy about 0.01% without Gd pins or after peak reactivity with Gd pins (within 0.1% before peak reactivity with Gd pins) and maximum pin powers about 1%.

The Feasibility Study of the Minimum-Shuffling Reloading Strategy for PWR

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The minimum-shuffling (MS) reloading strategy for PWRs is proposed as a fuel shuffling method to increase the load factor. In general, the full core shuffling is performed for PWRs, in which all fuel assemblies are once discharged and then reloaded with fresh fuel assemblies to make a next core. In the MS method, short intermediate shutdown just for fuel shuffling is performed. In this period, only highly burnt fuel assemblies are discharged and fresh fuel assemblies are inserted into the vacant positions. The other fuel assemblies are fixed during intermediate shutdown. This fuel reloading method can make the outage time shorter, and improvement of the load factor is expected. The MS method is applied to the 24 months cycle operation with intermediate shutdown. Loading patterns to satisfy various safety limitations are designed assuming the MS method. Economics analyses were carried out and it is confirmed that economics of the MS method is better than that of the conventional reloading strategy i.e. the full core shuffling.

Probability Approaching Method (PAM) and Its Application on Fuel Management Optimization

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Because fuel assemblies in this reactor are discharged normally after several fuel cycles, the refueling design optimization is in fact a multi-cycle problem. A typical multi-cycle problem consists of two kinds of simultaneous combined relations. One is the combined relationship of the locations where the fuel assemblies are placed at; the other is the combined relationship of fuel loading patterns among the relevant consecutive fuel cycles. This makes the problem very complex and difficult to solve.

In order to solve the multi-cycle optimization problems efficiently, a new method has been developed. By variable conversion, the multi-cycle optimization problem is de-coupled into a number of relatively independent mono-cycle issues. And the new problem becomes a non-linear programming problem with constraints.

For the non-linear programming problem with complex constraints, because of the high complicacy of the constraint, it is very difficult to know feasible region distribution clearly in the whole searching zone. This is just what makes it difficult to solve the problem. Usually, what we can do involves only:

- to know feasible region distribution roughly in the whole searching zone.
- to know feasible region distribution clearly in a local searching zone.

According to these facts, an algorithm for finding optimal solution based on probability theory is presented, namely probability approaching method (PAM). The complex constraint is converted to a rough probability condition by using a statistic method, then the non-linear programming problem is converted into a probability problem. By solving this probability problem, the probability distribution of optimal solution is obtained, which tells us the suitable searching area of next step. Then the probability condition converted from the constraint is updated by using new statistic area, and the probability problem is solved again. Iteration is executed and the optimal solution can be obtained at last. That is to say, the strict constraint is approached by a probability constraint, which is converged more and more with iterations. The result on simplified core model shows well effect of this new multi-cycle optimization scheme.

5A. Fuel/Core Design and Analysis

Session Organizer: Kord Smith (Studsvik-Scandpower).
Session Chair: Richard J. Cacciapouti (AREVA).

High Fuel Burn-Up and Nonproliferation in PWR-Type Reactor on the Basis of Modified Th-Fuel

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Neutronics-physical characteristics of the fuel lattice of a PWR-type reactor cooled by light water and by a mixture of light and heavy water have been analyzed. Th-fuel containing an essential amount of ^{231}Pa and ^{232}U is used, which allows an increase in fuel burn-up by a factor of 2-5 compared with that of traditional oxide uranium fuel with light water. It is important to underline that this is attained under the negative coolant density reactivity effect using cross sections of ^{231}Pa and ^{232}U from the updated JENDL-3.2 nuclear library. This radical increase of fuel burn-up is accompanied by a small change of reactivity during fuel irradiation (K_∞=1.1÷1.0), that favorably affects safety parameters of the reactor operation. A considerable percentage of ^{232}U in fuel, and consequently in U, is a strong barrier against the proliferation of such weapon nuclide as ^{233}U.

Over-Moderated MOX Fuel Assembly in a BWR Mixed Reload

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The use of MOX fuel in thermal reactors is now a mature technology. In several countries such technology has evolved at an industrial level and currently is a common practice to load a core partially with MOX fuel. Within local policies the use of plutonium fuel is part of a global nuclear fuel cycle strategy. In this work, the use of MOX fuel is presented as an alternative to reduce the nuclear waste from a nuclear power reactors, this is to recycle the spent fuel from the reactor will take us to the use of plutonium fuel. Currently exists several designs of MOX fuel assemblies, mainly, for PWR reactors and much less for BWR, these designs, in general uses the same mechanical design as the uranium fuel, moreover, they uses the same constrains and thermal limits as the uranium fuel, so is a common practice to match the reactivity of both kind of fuels, in all this the nuclear design of the fuel assembly is very important and is necessary to explore several alternatives, one of them is the over moderation of the fuel assembly as a way to improve the neutronic performance of the plutonium fuel, several results indicate that using this method is possible to have some advantages. The best result of several designs obtained by ININ are presented in this work.

Performance Comparison of Different Absorbent Materials in BWR Control Rods

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Nowadays there is a trend of increasing the cycle length of commercial operation in nuclear reactors. This demands to control a greater quantity of reactivity excess in this type of cycles.

In this work, a comparative study of the use of various absorbent materials in the control rods of a BWR is presented. The behavior of the different absorbent material used is accomplished only from the neutron point of view. The capability of each different material to control the reactor is analyzed by studying the required control rod density and the complexity to design the target control rod patterns (CRP).

The system GACRP was used to generate the target control rods patterns for an 18 months equilibrium cycle. This system is based on genetic algorithms computational technique. The found results show that the behavior of the dysprosium is acceptable and very similar to the boron one. In average, both require around 4.3% of control rod density,

while in Hf cases 5.1% is needed. Although the greatest impact of including different absorbent material is noted in cold condition, GACRP could obtain a CRP in any of the considered cases.

Enriched Gadolinium as Burnable Absorber for PWR

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This paper is a summary of a master of thesis work in reactor physics made by Ola Seveborn. The work was done at Vattenfall Bränsle AB and Ola was guided through the work by the corresponding author of this paper.

The results presented are calculations for Ringhals 3, which is a Westinhouse 3-loop PWR within the Vattenfall Group. The fuel is characterized by 17x17 assemblies of AFA type containing 3.80-3.95 w/o ^{235}U and 8 rods containing 2 w/o Gadolinium with an enrichment of 70 w/o ^{157}Gd. The calculations were performed with the Studsvik-Scandpower code package based on the CASMO-4 lattice code and the SIMULATE-3 nodal code.

The results are compared to the corresponding calculations for fuel with 5 w/o gadolinium with natural isotopic constitution. The depletion of the cores was done separately for the reference and enriched case.

The results show that the gains in average for the five cycles studied are about 70 EFPH per cycle. This is an effect of the lower gadolinium content needed. Also less parasitic absorption of enriched gadolinium in the end of the fuel life contributes to the increased cycle lengths. The abruptly increased reactivity and internal power peaking factor around 10 MWd/kgU do not affect the core design negatively.

The Pebble Bed Modular Reactor Layout and Neutronics Design of the Equilibrium Cycle

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The Pebble Bed Modular Reactor (PBMR) is a high-temperature helium-cooled, graphite moderated pebble bed reactor with a multi-pass fuelling scheme. The design, that displays the characteristics of Generation IV reactors, has reached a high level of maturity. Excess reactivity is limited by continuous refueling while adequate passive heat removal ensures an inherent safe design with no event with significant fission product release being possible.

The core neutronic design for the 400 MWth PBMR is performed by the VSOP99 code system. The design is for an annular core with an outer diameter of 3.7 m and an inner diameter of 2 m shaped by the fixed central reflector. The effective cylindrical core height is 11 m.

For the equilibrium core VSOP results show that the fuel sphere powers (maximum 2.7 kW) and operational temperatures (<1100 ˚C) fulfil the design criteria. Adequate reactivity control and long-term cold shutdown are provided by two separate and diverse systems while the overall negative reactivity temperature coefficient is illustrated over the total operational range.

Neutronic Modeling for a Gas-Cooled Fast Reactor Assuming Coated Fuel Particles

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This paper focuses on the modeling of Gas Cooled Fast Reactor with the APOLLO2 code. It aims to estimate the APOLLO2 code accuracy in range of fast neutron reactors. The study consists in the comparison of calculations with APOLLO2 code, and TRIPOLI4 Monte Carlo code.

The two level PIJ/SN APOLLO2 scheme proposed provides promising results. Considering different core and cell configurations, calculations indicate a low residual reactivity discrepancy [300 - 625 pcm] between TRIPOLI4 and APOLLO2.

An isotopic decomposition pointed out that a mixture of isotopes with resonance in the same energy interval degrades the self-shielding process. Two new developments in self-shielding have been tested. The first consists in treating the resonant mixture as a unique entity (mixture self-shielding). The second model takes into account the transfer self-shielding. The first model mitigates the problem of resonance overlapping in the slowing down range. The second method corrects the transfers to high energy. The transfer self-shielding of ^{238}U has an important effect since it permits to recover about 150 pcm on reactivity.

To conclude, k_{eff} discrepancies are satisfactory enough to use this scheme for core design calculation, but errors remain in the self-shielding process of the 2 major isotopes (^{238}U, ^{239}Pu). Furthermore, the 33-group diffusion calculation is a compromise in terms of accuracy and computation time for GFR dimensional studies.

Methodology for a Large Gas-Cooled Fast Reactor Core Design and Associated Neutronic Uncertainties

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After 600 MWth cores previous studies performed on, a feasibility study of a 2400 MWth gas-cooled fast reactor using neutronic and thermo-hydraulic constraints has been performed. Considering larger cores do not imply any change in the safety approach but relax some of the design constraints on the fuel technology, on the fuel residence time and on the power density. These changes allow an enhanced economy competitiveness. The reference core has a 100 MW/m^3 power density, is based on a dispersed fuel with a fuel-to-matrix volume ratio of 50/50 and achieves a breeding ratio of

1.0 without fertile blankets. This concept possesses enhanced safety features due to a large Doppler effect owing to the presence of carbon in the SiC matrix. The possibility to remove the decay heat out of the core by natural circulation of the gas under a minimum back-up pressure is kept by limiting the core pressure drop. Numerical validation of deterministic calculations by comparisons with Monte-Carlo results are presented and uncertainties due to nuclear data are quantified. It is shown that the specificities of gas-cooled fast reactors keep bias and uncertainties within reasonably limits which are sufficient for current pre-design studies.

Optimal Moderation in the Pebble-Bed Reactor for Enhanced Passive Safety and Improved Fuel Utilization

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The concept of optimal moderation is introduced as a reactor configuration in which any change in the fuel-to-moderator ratio causes a reduction in reactivity. It is shown that such a configuration cannot persist over time in a reactor with stationary fuel, but can do so in a reactor with moving fuel, such as a pebble-bed reactor. It is further shown by the example of pebble-bed versions of the proposed Next Generation Nuclear Plant that such optimal moderation in gas-cooled reactors leads to significant advantages in response to water ingress, fuel utilization, and resistance to nuclear weapons proliferation.

Fuel Design and Core Layout for a Gas Cooled Fast Reactor

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The key goal for the Gas Cooled Fast Reactor is sustainability, with sel-breeding capability and full recycling of trans-uranics. Blankets are minimized to improve proliferation resistance. In this paper two core concepts using two types of fuel elements are presented for a 2400 MWth GCFR. One concept uses coated particles, redesigned from TRISO (HTR) CPs, and the other concept uses hollow fuel spheres, an innovative type of fuel element, featuring a hollow shell of fuel with ceramic cladding. Both cores have direct cooling by helium flowing through the beds of fuel elements and a high outlet temperature. Discharge LWR Pu is used as fissile material. A rudimentary core layout is presented for both concepts: the hollow fuel spheres are packed in beds, the CPs are arranged in annular cilinders. The results indicate that a self-sustaining core can be achieved using CPs or hollow fuel spheres, without the use of blankets. The core with CPs requires a slightly larger fuel inventory. The core with hollow fuel spheres can be designed to have a positive reactivity swing with burnup. The fuel temperature coefficient is always negative. Both core concepts are well within the tentative limits set within the Generation IV GCFR framework.

GT-MHR Core Modeling: From Reference Modeling Definition in Monte-Carlo Code to Calculation Scheme Validation

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Today, the HTGR (High Temperature Gas cooled Reactor) appears as a promising reactor concept for the next generation of nuclear power applications. The CEA, in collaboration with FRAMATOME, is developing a core modeling dedicated to the prismatic block-type reactor. This calculation scheme is based on a usual two-steps Transport – Diffusion approach. It will have to serve for design studies and industrial calculations as well as for best estimate and reference calculations.

Due to the lack of usable experimental results, the reliability of the code system used for the design studies is essentially based on results obtained with reference calculations. Such reference calculations are usually obtained with Monte-Carlo codes. These codes allow modeling the core geometry and its characteristics without any physical assumptions. However, firstly the Monte-Carlo method cannot be considered today as a reference for validating core burnup calculation and secondly, in HTGR, the fuel is in a form of dispersed particles (coated fuel particles, embedded in a graphite matrix). It imposes the treatment of stochastic geometries in Monte-Carlo calculations and may question the principle of the absolutely unbreakable reference that constitutes the Monte-Carlo methods. Indeed, assumptions must be done for the geometrical description of the stochastic medium in Monte-Carlo codes.

The purpose of the paper is to evaluate the physical impact of these assumptions in the CEA-FRAMATOME framework that consists in qualifying and validating the deterministic modeling of the GT-MHR loaded with uranium fuel. Mastering the uncertainties of the modeling in Monte-Carlo codes allows us to use these calculations for the validation of the computational tools used for conceptual studies. This is the purpose of the second part of the paper which is dedicated to the validation of the computational tools in a simplified geometry. This simplified geometry is a 2D core modeling which is representative of all the physical effects generally encountered in this type of core (annular core configuration with a strong coupling between fuel element and reflector).

Modeling of HTRs with Monte Carlo: Sensitivity Due to Different Isotopic Fuel Composition

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The gas turbine modular helium-cooled reactor (GT-MHR) is a potential candidate for the maximum plutonium destruction in once-through cycle. A particular feature of GT-MHR is that its refractory coated fuel (TRISO particles) is supposed to provide an impermeable barrier to the release of fission products and, at the same time, to resist very deep burn-up rates (more than 90% for ^{239}Pu).

In this work we performed detailed Monte Carlo simulations of the GT-MHR operation by loading with different plutonium fuel vectors: plutonium from military applications, plutonium from LWR and RBMK spent nuclear fuel. The comparison of the main GT-MHR performance parameters: k_{eff} eigenvalues, the length of the fuel cycle, neutron characteristics and the evolution of fuel composition in particular were obtained. We show that the performance of GT-MHR may be considerably influenced by the plutonium isotopic composition vector used as initial fuel material. This is the first time when incineration possibility of the RBMK-1500 based plutonium isotopic composition was tested using high temperature reactor. Finally, the propagation of statistical errors for the fuel burn-up was examined in detail showing no uncertainty accumulation.

Low-Conversion Ratio Gas-Cooled Fast Reactors

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The neutronic feasibility and performance of gas-cooled fast burner (GFR) cores designed for low conversion ratio (0.86 to 0.00) have been investigated. Two distinctly different fuel types were evaluated. These are a block-type fuel assembly using dispersion fuel in a low neutron absorbing, neutron moderating SiC matrix and a pin-type fuel assembly using solid solution fuel clad with neutron absorbing Nb-1Zr.

The results of these calculations show several trends with potential safety implications. Very low conversion ratios (CRs) result in large burnup reactivity losses. The delayed neutron fraction and prompt neutron lifetime decrease with conversion ratio. The magnitude of the Doppler reactivity coefficient is reduced at very low conversion ratios and is near zero for the pin fuel. For the block fuel, the Doppler reactivity coefficient has a large negative value for CR=0, which may be a significant safety advantage for this type of fuel. As the conversion ratio decreases, the helium void worth remains roughly constant for the block fuel, but increases significantly for the pin fuel.

The results for the GFR designs compare favorably with those for a sodium-cooled fast reactor (SFR). Parametric studies were performed to evaluate the impact of a number of design choices, which had relatively little effect on the performance of the GFR.

Possibility to Use Different Fuel Cycles in GT-MHR

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The GT-MHR reactor core is characterized by flexibility of neutronic characteristics at the given average power density and fixed geometrical dimensions of the reactor core. Such flexibility makes it possible to start the reactor operation with one fuel, and then to turn to another type of fuel without changes of main reactor elements: fuel block design, core and reflector dimensions, number control rods and their positions etc.

Preliminary analysis reindicates the commercial viability of the GT-MHR concept, part of which is due to its ability to accommodate different fuel types and cycles. This paper presents the results of studies of the neutronic characteristics of GT-MHR cores using different types of fuel (low- and highly-enriched uranium, pure weapons or civil plutonium, MOX fuel).

Comparison of different fuel cycles is carried out for a three-batch refueling option with respect to following characteristics: discharged fuel burnup, reactivity change during one partial cycle of fuel burnup, consumption of fissile isotopes per unit of produced energy, power distribution, reactivity effects, control rods worth.

It is shown, that the considered options of fuel loads provide the three-year core lifetime (with account for the load factor ~0,8) without changes of core design, number and design of control rods at transition from one fuel type to another.

5C. Critical and Subcritical Experiments

Session Organizers: Russell D. Mosteller (LANL), Hironobu Unesaki (Kyoto University).
Session Chairs: Russell D. Mosteller (LANL), Alain Santamarina (CEA Cadarache).

Criticality Analysis of Highly Enriched Uranium/Thorium Fueled Thermal Spectrum Cores of Kyoto University Critical Assembly

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A series of critical experiments on thermal spectrum cores containing highly enriched uranium and thorium have been performed at Kyoto University Critical Assembly (KUCA) of Research Reactor Institute, Kyoto University, Japan. Seven critical cores with systematic variation on neutron spectrum and ^{232}Th/^{235}U ratio have been constructed in the experiment. Analysis of criticality (k-effective) have been performed using continuous energy Monte Carlo code MVP and various nuclear data libraries such as JENDL-3.2, JENDL-3.3, ENDF/B-VI.8 and JEFF3.0.

It has been found that the large overestimation of k-effective observed for JENDL-3.2 is significantly reduced by the use of JENDL-3.3. However, C/E values by JENDL-3.3 range from 1.007 to 1.009 and are significantly larger than C/E values of cores without thorium. Considerable spread among the ^{232}Th cross sections exist and have been shown to have considerable impact on nuclear characteristics of thorium fueled thermal systems. Among the nuclear data libraries considered in this study, ENDF/B-VI.8 showed the best results in terms of criticality prediction.

Analysis of Criticality Change with Time for MOX Cores

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In plutonium-uranium mixed oxide (MOX) fuel, the composition changes with time due to the decay of ^{241}Pu (half life of 14.35 y) and build up of ^{241}Am. This change reduces the criticality of a MOX core with time because of the decrease in fissions and the increase in neutron captures. At the Japan Atomic Energy Research Institute (JAERI), a series of critical experiments for MOX cores had been performed, and criticality data as a function of time were obtained for about 7 years to investigate the effect of fuel composition change. We have analyzed the criticality change for those cores with a Monte Carlo code, MVP, employing the Japanese Evaluated Nuclear Data Libraries, JENDL-3.2 and 3.3.

The effective multiplication factors for the TCA critical cores showed the dependence on time, that is, they increased with time. This indicates that there exists some errors in cross sections of ^{241}Am and/or ^{241}Pu.

Subcritical Experiments in Uranium-Fueled Core with Central Test Zone of Tungsten

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To study the basic core characteristics of accelerator-driven systems (ADS), subcritical experiments were carried out in a uranium-fueled core FCA-XXI-1 at the FCA. Subcritical reactivity and axial distributions of ^{235}U-fission rate were measured in subcritical configurations with the central test zone of tungsten for simulating a target region of ADS. An external neutron source of ^{252}Cf was used to drive the subcritical core.

The deterministic calculations by using a conventional fast reactor analysis code system with the JENDL-3.2 nuclear data file resulted in an overestimation of 1~2% for the effective multiplication factor k_{eff} of critical configuration, while the precisely modeled Monte Carlo calculation showed a good accuracy of 0.2%. The subcritical reactivity was measured down to -5%Δk by the modified source multiplication method. The measured reactivity values by out-of-core

detectors showed little dependence of source position and agreed with the calculated values within 5%, but those by in-core detectors still showed a spatial dependence on the positions of external source and detectors. The measured distribution of ^{235}U fission rate for each subcritical configuration was well calculated by adjusting effective production cross-sections so that the calculated k_{eff}-value of the reference critical configuration should be consistent with the experimental one.

Re-Evaluation of SEFOR Doppler Experiments and Analyses with JNC and ERANOS Systems

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The SEFOR (South-West Experimental Fast Oxide Reactor) static Doppler reactivity experiments have been re-evaluated. The re-evaluation was carried out on the experiments performed at power levels up to 20MW, starting from a review of raw data of the Doppler reactivity and fuel temperature.

Investigations were carried out on various parameters, such as fuel thermal conductivity and gap conductance, or weighting function on the temperature distribution in the core. Experimental uncertainties on some parameters were also re-evaluated.

Based on JNC calculation with the most recent thermal conductivity correlation and an exact weighting, quite good C/E values were obtained at higher powers.

The Doppler constant was also re-evaluated from the Doppler reactivity at 20MW. Resulting constants are 3 or 5% larger than the original GE evaluation. The increase is mainly attributed to the update of the fuel thermal conductivity correlation.

The new values look more reasonable than the recommended HEDL evaluation in that C/E values based on both JNC and CEA do not depend on the core type tested in the experiment.

In addition, the experimental uncertainty is significantly reduced from 12% to 6%, and this is of the utmost importance according to current accuracy requirements.

The TRADE Source Multiplication Experiments

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This paper presents the results of the first sub-critical measurement campaign performed for the TRADE (TRiga Accelerator Driven Experiment) program in late 2003. The TRIGA reactor at the Casaccia Center of ENEA was studied in a number of configurations ranging from k_{eff} approximately 0.993 to 0.93. Over a two week period, starting from a single reference configuration, fuel elements were removed, control rods were moved outward (required for the eventual TRADE experiments), and fission chambers were inserted. In all configurations, sub-critical counts were taken.
The results of these measurements are given in this paper. First estimates of reactivity in all the states are presented, and our first attempts at correcting the reactivity based on calculations of source importance and detector efficiency– so called MSM factors—are also presented.

Study of the Influence of Source Type in the Kinetics Measurements in a Subcritical System

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The interest in Accelerator Driven Systems (ADS) has increased the attention for the measurements of kinetic parameters in subcritical assemblies driven by an external source. A pulsed source is time-dependent, thus we have reformulated the corresponding methods based on the fluctuation measurements. Because the time distribution of the neutron population is used to infer the reactivity of the subcritical system, we must also consider the time dependent propagation of the external source in space and energy. The time distribution of a pulsed source give us information not only about the fission time distribution but about the reflector influence in the reactor system and its material composition.

One of the goals of the MUSE 4 program at the Cadarache Center of CEA (France) is to investigate methods to measure and monitor the reactivity in an ADS. A deuterium accelerator (GENEPI) with a deuterium and a tritium target has been coupled to a subcritical reactor (MASURCA). To study the influence of a source driven subricitical system we have performed measurements at the same reactivity with 3 different type of sources: Cf-252 (spontaneous source), deuterium-deuterium and deuterium-tritium pulsed sources. We show some results from these measurements.

The MUSE4 Pulsed Neutron Source Experiments

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The latest results of the Pulsed Neutron Source experiments of the MUSE4 project, performed in the framework of the European 5FP, are described. This includes three main symmetric configurations from near criticality to K_{eff}=0.95, each in two variants with 150pcm difference, plus several configurations with safety rods inserted.

Near critical loadings are easily described by point kinetics. However largely subcritical configurations require more complex description. A model based on the Avery theory of coupled systems has been successfully compared with the experimental data.

Two analysis methods, one based on the counting rate time decay shape of the couple systems model and the other on the Sjöstrand ratio between the prompt and delayed neutrons integrals, are presented and compared.

The two methods are complementary. The first one obtain its information from the asymptotic behavior of the reactor, but is sensible to the kinetic parameters and reaches “only” precisions of 5%. The second one is based on the neutron multiplication, insensitive to kinetic parameters but it is probably affected by spatial and local spectral effects and reaches precisions between 0.5 and 1%.

Both methods provide excellent results for configurations from near criticality to K_{eff} = 0.97. There are, however, small differences at K_{eff} = 0.95 of about 15% that are still under investigation.

Session Organizers: Russell D. Mosteller (LANL), Hironobu Unesaki (Kyoto University).
Session Chairs: Russell D. Mosteller (LANL), Alain Santamarina (CEA Cadarache).

Reactivity Measurements and Neutron Spectroscopy in the MUSE-4 Experiment

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Reactivity measurement gives access to an important parameter in reactor physics, which can be achieved by many experimental techniques. This paper describes our current efforts to develop and test a method which makes use of the Pulsed Neutron Source technique. To do so, the MASURCA fast neutron reactor was coupled to the pulsed neutron generator GENEPI. For various subcritical configurations, the decay of the neutron population which follows a neutron burst was recorded using in-core fission chambers. The analysis relies on the distribution of time intervals between fission events belonging to the same fission chain. An excellent agreement is found between the measured reactivities and the expected ones, from a near criticality configuration down to very deep subcritical levels. The second part of the paper is devoted to the measurement of the neutron energy distribution. A proportional counter was used to measure the energy deposition of the neutrons in the ^{3}He active gas. A calibration of the counter response allows us to reconstruct the neutron flux at the detector location. A reasonable agreement is found with a Monte Carlo prediction. This gives us a direct test of the stochastic approach to the neutron transport.

Some Experimental Results from the Last Phases of the MUSE Program

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The fourth phase of the MUSE program, the MUSE-4 experiments, started on October 2000 within the frame of the 5th Framework Program. Based on a parametric approach and the use of a well known external source, in terms of intensity and neutron energy, provided by the GENEPI (Générateur de Neutrons pulses Intense) neutron generator [1], the major objectives of the MUSE-4 experiments are to investigate the physics of externally-driven subcritical multiplying media and to study the possibility to infer the reactivity levels of such systems without the need of a critical configuration.

In the last three years we have progressed from critical reference measurements to measurements at k_{eff}~0.995, 0.97, and finally 0.95. Most of the measures have been performed with DD and DT sources. In this paper, we give an overview of the experiment progress and a sample of the results that were obtained up to now. As concerns kinetics measurement, the outcomes of on going analysis is presented. A comparison between the various methods used for the reactivity determination is notably completed.

5D. Reactor Analysis Methods

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Elmer Lewis (Northwestern University).

Reactor Core Simulations in Canada

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This review will address the current simulation flow-chart currently used for reactor-physics simulations in the Canadian industry. The neutron behaviour in heavy-water moderated power reactors is quite different from that in other power reactors, thus the core physics approximations are somewhat different. Some codes used are particular to the context of heavy-water reactors, and the paper focuses on this aspect. The paper also shows simulations involving new design features of the Advanced CANDU Reactor (ACR), and provides insight into future development, expected in the coming years.

A New Method for the Treatment of Local Strong Heterogeneities and Its Application to the Phebus Experimental Facility

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This paper describes a new approach to take into account strong localized heterogeneities in transport core calculations and its application to a benchmark model of the PHEBUS reactor facility at CEA Cadarache.

Our recently-developed decomposition method allows to distinguish, within a full core calculation, several sub-domains where appropriate transport equation and resolution techniques can be used. Moreover, in each sub-domain the mesh size is independently refined to take advantage of the selected solution method.

This MultiMethod/MultiDomain decomposition method relies on two kinds of transport methods. The first one, a discrete ordinates (S_n) nodal and characteristics method, is employed with highly refined meshes for the treatment of strong local heterogeneities. The second one, the Variational Nodal Method, is a powerful method for the treatment of the surrounding core, with a very coarse mesh discretization.

To show the efficiency of these techniques, a benchmark model of the PHEBUS reactor, in which a central experimental fuel element require a finer mesh description than the surrounding core, has been defined.

Several calculations have been done on different grids. The most important result is that our method allows us to reduce the CPU time by a factor of 3, while preserving the accuracy of the main neutronic indicators: ≤ 50 pcm for reactivity, ≤ 1% for the flux error and ≤ 0.50 % for the core-to-bundle power coupling coefficient.

Mixed-Hybrid Methods for the Linear Transport Equation

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Mixed-hybrid methods generalize the existing mixed and hybrid methods combining their attractive features. A significant achievement is that not only odd- but also even-parity P_N approximations are used for the angular discretization. This leads to a promising enclosing property in our numerical tests. Furthermore, we establish flux and current interface continuity properties from the Rumyantsev conditions, for which we outline a new derivation.

5D. Reactor Analysis Methods

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Elmer Lewis (Northwestern University).

Adaptive Solution of the Multigroup Diffusion Equation on Irregular Structured Grids using a Conforming Finite Element Method Formulation

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In this paper, a method for performing spatially adaptive computations in the framework of multigroup diffusion on 2-D and 3-D Cartesian grids is investigated.

The numerical error, intrinsic to any computer simulation of physical phenomena, is monitored through an *a posteriori* error estimator. In *a posteriori* analysis, the computed solution itself is used to assess the accuracy. By efficiently estimating the spatial error, the entire computational process is controlled through successively adapted grids.

Our analysis is based on a finite element solution of the diffusion equation. Bilinear test functions are used. The derived *a posteriori* error estimator is therefore based on the Hessian of the numerical solution.

The Variational Nodal Method in *R-Z* Geometry

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The variational nodal method contained in the Argonne National Laboratory code VARIANT is generalized to include *R-Z* geometry. Spherical harmonic trial functions are used in angle, and polynomials in space. The nodal volumes correspond to toroids, with rectangular cross sections, except along centerline where they are cylinders. The *R-Z* response matrix equations are solved using the iterative methods already contained in VARIANT. Results are given for both a one-group fixed source and a two-group eigenvalue problem.

Solving the Neutron Diffusion Equation on Combinatorial Geometry Computational Cells for Reactor Physics Calculations

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An approach is developed for solving the neutron diffusion equation on combinatorial geometry computational cells, that is computational cells composed by combinatorial operations involving simple-shaped component cells. The only constraint on the component cells from which the combinatorial cells are assembled is that they possess a legitimate discretization of the underlying diffusion equation. We use the Finite Difference (FD) approximation of the x,y-geometry diffusion equation in this work. Performing the same combinatorial operations involved in composing the combinatorial cell on these discrete-variable equations yields equations that employ new discrete variables defined only on the combinatorial cell's volume and faces. The only approximation involved in this process, beyond the truncation error committed in discretizing the diffusion equation over each component cell, is a consistent-order Legendre series expansion. Preliminary results for simple configurations establish the accuracy of the solution to the combinatorial geometry solution compared to straight FD as the system dimensions decrease. Furthermore numerical results validate the consistent Legendre-series expansion order by illustrating the second order accuracy of the combinatorial geometry solution, the same as standard FD. Nevertheless the magnitude of the error for the new approach is larger than FD's since it incorporates the additional truncated series approximation.

5D. Reactor Analysis Methods

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Elmer Lewis (Northwestern University).

Development of Hybrid Core Calculation System using 2-D Full-core Heterogeneous Transport Calculation and 3-D Advanced Nodal Calculation

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This paper presents the Hybrid Core Calculation System which is a very rigorous but a practical calculation system applicable to best estimate core design calculations taking advantage of the recent remarkable progress of computers. The basic idea of this system is to generate the correction factors for assembly homogenized cross sections, discontinuity factors, etc. by comparing the CASMO-4 and SIMULATE-3 2-D core calculation results under the consistent calculation condition and then apply them for SIMULATE-3 3-D calculation. The CASMO-4 2-D heterogeneous core calculation is performed for each depletion step with the core conditions previously determined by ordinary SIMULATE-3 core calculation to avoid time consuming iterative calculations searching for the critical boron concentrations while treating the thermal hydraulic feedback. The final SIMULATE-3 3-D calculation using the correction factors is performed with iterative calculations searching for the critical boron concentrations while treating the thermal hydraulic feedback.

A First-Order Spherical Harmonics Formulation Compatible with the Variational Nodal Method

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A spherical harmonics method based upon the first-order transport equation is formulated and implemented into VARIANT [1,2], a variational nodal transport code developed at Argonne National Laboratory. The spatial domain is split into hybrid finite elements, called nodes, where orthogonal polynomial spatial trial functions are used within each node and spatial Lagrange multipliers are used along the node boundaries. The internal angular approximation utilizes a complete odd-order set of spherical harmonics. Along the nodal boundaries, even and odd-order Rumyantsev interface conditions are combined with the spatial Lagrange multipliers to couple the nodes together. The new method is implemented in Cartesian x-y geometry and used to solve a fixed source benchmark problem.

Investigating the Use of 3-D Deterministic Transport for Core Safety Analysis

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A project is underway at the Idaho National Engineering and Environmental Laboratory (INEEL) to demonstrate the feasibility of using a three-dimensional multi-group deterministic neutron transport code to perform global (core-wide) criticality, flux, and depletion calculations for safety analysis of a nuclear reactor for the INEEL's Advanced Test Reactor (ATR). The ATR is the world's premier materials test reactor. Its 250 MW, beryllium-reflected core contains large flux trap and pressurized loop facilities for conducting accelerated irradiations of fuel and structural materials.

The neutron transport code Attila solves the discrete ordinates (S_n) equations on an unstructured tetrahedral mesh. To create the mesh, the geometry of a core plane is modeled using Solidworks. The Discontinuous Galerkin Finite Element method used in Attila provides a higher order solution of flux and current. The current ATR model, in which the 19-plate fuel elements are homogenized into 3 radial sections, contains about three million mesh elements. Each element has four spatial, four energy; and 24 angular (P_1 approximation) unknowns for a total of over a billion in the problem. Only recently can such a computationally intensive problem be solved on available computers.

Comparisons of Attila results to experimental data indicate the feasibility of using a three-dimensional deterministic transport code for core safety analysis.

Feasibility and Configuration of a Mixed Spectrum Supercritical Water Reactor

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Assessment of the feasibility issues and fuel cycle performance of the mixed spectrum supercritical water reactor (MS^2) has been performed. Through pitch-to-diameter (P/D) ratio sensitivity studies, the feasible P/D ratio windows for the thermal zone of the MS^2 core are sought for stainless steel and Ni-based claddings, which is large enough to provide adequate reactivity and small enough to provide an adequate heat transfer coefficient.

By choosing the P/D ratio within the feasible windows, the MS^2 core has been designed and evaluated from safety and actinide management viewpoints. This core has negative moderator temperature reactivity coefficient and the shutdown margin is 1.6% $\Delta\rho$. By recycling Pu, Np and Am in the fast zone, the MS^2 concept is capable of keeping Pu, Np and Am in the reactor fuel cycle and thus eliminating them from the disposed nuclear waste.

Experimental Study on Reduced Moderation BWR with Advanced Recycle System (BARS)

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Experimental study has been done for reduced-moderation spectrum boiling water reactor named BARS (BWR with Advanced Recycle System). The advanced recycle system means the combination of reduced moderation BWR, dry reprocessing and vibro-packing of MOX fuel fabrication. This system simplifies reprocessing and MOX fuel fabricating process and reduces related backend cost. The reduced moderation spectrum condition is obtained through triangular tight fuel rod lattice configuration and higher void fraction. This feature causes some design problem on nuclear and thermal hydraulic area. According to resolve the problem, critical assembly experiment and thermal hydraulic test have been done under the tight lattice configuration. The critical assembly experiment for triangular tight lattice has been done at Nuclear Critical Assembly (NCA) in Toshiba. Experimental method based on modified conversion ratio has been adopted to evaluate the void reactivity effect. The measured void coefficient for tight lattice agreed with the Monte Carlo calculation. Visual test and high-pressure thermal hydraulic test have been done as the thermal hydraulic test. Visual test has indicated the flow behavior for BARS lattice is almost the same as current BWR. The high-pressure thermal hydraulic test has indicated the applicability of modified Arai's correlation to the BARS lattice.

A Core Design for a Single Fuel Enrichment in a Self-Sustaining Lead-Cooled Reactor

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Reactor physics studies have been done to achieve single fuel enrichment (SFE) in a 900 MWth lead-cooled breakeven reactor with a burnup reactivity swing smaller than the β_{eff} value. For the single fuel enrichment in an identical fuel rod type, new fuel assembly designs have been introduced: a combination of B_4C burnable absorber rods, neutron streaming tubes, and moderator rods are utilized to control power distribution. The burnable absorber rods are designed to have top and bottom cutback regions to reduce the peak fast fluence and enhance overall boron depletion rate. An 18-month cycle core has been designed and its various physics characteristics are analyzed. Additionally, subchannel thermal-hydraulic analyses have been performed for the peak power assembly.

The analyses have shown that the proposed SFE design schemes are effective ways to design a high-performance breakeven core with an identical rod type. With the proposed approaches, separation of transuranics from the spent fuel is not required. Thus, related fuel cycle can be highly proliferation-resistant and the reprocessing fuel fabrication would be significantly simplified.

Preliminary Neutronics Design Studies of a Lead Cooled, Small Modular Reactor

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This paper explores the feasibility of a small, lead-cooled modular reactor. The main design criteria that must be satisfied are long fuel lifetime, natural convection heat transport, semi-autonomous control, and small unit size. These design goals inherently require a derated power level, low burnup swing, and open pin bundle design. Parametric neutronics studies were performed to determine the minimum core size required to satisfy these design criteria for several different fuel pin configurations.

Targeting a 20 year lifetime and 25 MWt power level, the natural circulation design objective was easily met. The semi-autonomous control design objective is also possible, but modifications to the core geometry are likely necessary to facilitate better reactivity feedback from geometrical expansion. The transportability design objective is the only point of contention due to the large size and weight of the core. More precise limits on the thermal hydraulic assessment of the reactor design can help reduce the core size and fuel loading.

Physics and Safety Studies of a Low Conversion Ratio Sodium Cooled Fast Reactor

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This paper explores the feasibility of a compact fast burner reactor that can achieve a low transuranic conversion ratio. The major design option considered is the reduction of fissile breeding by the removal of fertile material from the fast reactor system. Reductions in the fuel pin diameter and thus fuel loading were employed to remove fertile material. Reactor performance parameters and reactivity coefficients were evaluated for a compact core design with a targeted conversion ratio of 0.25. To assess the safety implications, a detailed transient analysis model was employed using the SAS4A/SASSYS-1 computer code.

A series of calculations was performed to assess the behavior of the reactor and plant in an unprotected loss-of-flow accident (ULOF). A parametric study was also carried out using increasingly conservative modeling assumptions. The computational results show that for nominal, best-estimate analysis assumptions and input data, the low conversion ratio reactor design responds to the ULOF with a very high level of self-protection. Both short-term and long-term quasi-equilibrium reactor conditions predicted in the analysis indicate very large margins of safety.

Preliminary Neutronics Design Studies for a 400 MWt STAR-LM

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Neutronics design studies for a 400 MWt high temperature fast reactor are being performed, utilizing lead coolant, transuranic (TRU) nitride fuel, and HT-9 structural material. Under the main design constraints of long fuel lifetime, natural convection heat transport, semi-autonomous control, and small unit size, parametric studies were performed to maximize the discharge burnup and minimize the burnup reactivity swing. Based on the results of these parametric studies, two point designs were developed for a single-batch once-through fuel cycle; one is a 15 full power year cycle

design with core volume of 9.5 cubic meters, and the other is a 12 full power year cycle design with core volume of 7.4 cubic meters. For these two point designs, fuel cycle analyses and reactivity feedback coefficients calculations were performed. The 9.5 cubic meter design achieved an average discharge burnup of 83 MWd/kg with a maximum reactivity change over the lifetime of 0.6%. The peak fast fluence was well within the fast fluence limit of HT9, and both average and peak power densities were well below the estimated limit for natural circulation. The performances of the 7.4 cubic meter design were slightly inferior to this design. To enhance the passive safety characteristics, however, further design improvements need to be made to reduce the coolant density coefficient and to increase the radial expansion coefficient.

PDS-XADS LBE and Gas-Cooled Concepts: Neutronic Parameters Comparison

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Though the problem of the closure of the nuclear fuel cycle is not completely solved, partitioning and transmutation may eliminate the largest part of the long-lived nuclear waste provided that a neutron flux with adequate energy and intensity can be employed. In this framework, the Accelerator Driven System (ADS), coupling a high energy, high current proton accelerator with a subcritical reactor has a remarkable potential, allowing the minimisation of the high level waste while operating in a safe manner.

Within the European Fifth Framework Program the Preliminary Design Studies of an eXperimental Accelerator Driven System (PDS-XADS) are focussed on options employing Lead-Bismuth Eutectic (LBE) and helium gas coolants. Two of the options employ 80 MWth subcritical cores, which are driven by a 600 MeV proton beam with a maximum current of 6 mA, the proton beam impinging on a windowless LBE target. The current comparison of the two systems in terms of burnup dependent static and kinetic subcriticality parameters exhibits a much larger transport effect in the Gas-Cooled XADS, reflecting the strong anisotropy of scattering in the low-density regions. Generally speaking, the Gas-Cooled XADS is more difficult to calculate, the overall neutron balance being determined by enhanced core/reflector interface effects.

Study of Accelerator Transient on ADS Operation using TRACY (Transient Experiment Critical Facility)

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The experiments simulated the dynamic behavior of ADS were performed by using TRACY (the transient experiment critical facility) and Pulsatoron (the pulse neutron generator). The experiments for three ADS operation patterns with two subcritical cores were conducted and the time responses of the three neutron detectors were measured. The results by those experiments clearly show the dynamic behavior of ADS for neutrons and/or reactivity insertion. Although the experiments were analyzed by a one-point kinetics code, one-point kinetics calculation was not sufficient to reproduce the measured time responses of the detectors. The experiments were also analyzed by a modified method combined with a one-point kinetics code and a continues energy Monte-Carlo code. The result by the modified method agreed well with the measured responses of all the three detectors. The present experimental and analysis works lead to supply the benchmark data for studying the dynamic behavior of ADS and to provide an advanced dynamic code combined with the one-point kinetics code and the continuous Monte-Carlo code.

Activities of Working Party on "Subcritical Core of Accelerator-Driven System" under the Research Committee on Reactor Physics of AESJ and JAERI

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The Research Committee on Reactor Physics under the Atomic Energy Society of Japan and the Japan Atomic Energy Research Institute organized the working party (ADS-WP) on "Subcritical Core of Accelerator-Driven System". The ADS-WP investigated reactor physics of subcriticality from the viewpoint of the accelerator driven system (ADS) since subcriticality has been almost studied from the viewpoint of critical safety. The working party was set in July 2001 and it worked for two years. The activities of the ADS-WP are (Work-I) theory of subcriticality, (Work-II) benchmark of subcritical core, (Work-III) setting of subcriticality level of ADS and (Work-IV) monitoring of subcriticality. These activities clarified about the important issues related to the subcriticality or the subcritical core from the wide ranges of theory, analysis, calculation, design and monitoring for ADS. The activities were already summarized and the report will be published in March 2004.

Analysis of Boron Dilution Transients in PWRs

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A study has been carried out with PARCS/RELAP5 to understand the consequences of hypothetical boron dilution events in pressurized water reactors. The scenarios of concern start with a small-break loss-of-coolant accident. If the event leads to boiling in the core and then the loss of natural circulation, a boron-free condensate can accumulate in the cold leg. The dilution event happens when natural circulation is re-established or a reactor coolant pump (RCP) is restarted in violation of operating procedures. This event is of particular concern in B&W reactors with a lowered-loop design and is a Generic Safety Issue for the U.S. Nuclear Regulatory Commission. The results of calculations with the reestablishment of natural circulation show that there is no unacceptable fuel damage. This is determined by calculating the maximum fuel pellet enthalpy, based on the three-dimensional model, and comparing it with the criterion for damage. The calculation is based on a model of a B&W reactor at beginning of the fuel cycle. If an RCP is restarted, unacceptable fuel damage may be possible in plants with sufficiently large volumes of boron-free condensate in the cold leg.

Uncertainty Evaluation of the Results of the MSLB Benchmark by CIAU Methodology

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The purpose of the presented research in this paper was to perform an uncertainty evaluation of a Main Steam Line Break (MSLB) transient scenario occurring in a PWR applying the CIAU-TN methodology (Code with - the capability - of Internal Assessment of Uncertainty for Thermal-hydraulics/Neutronics coupled codes).

The work has been carried out within a technical cooperation between Penn State University and University of Pisa where the CIAU-TN methodology has been developed. Two main objectives have been established. First, to supply the uncertainty evaluation to the results of the OECD/NRC PWR MSLB Benchmark. Upper and lower continuous limits have been predicted for the trends of the axial and the radial peaking factors of spatial power distribution and for the core power history. The second aim was to enlarge the CIAU-TN database that, currently, includes the uncertainty values derived by the analysis of the results of OECD/NRC BWR TT Benchmark.

Improving the Computation Efficiency of COBRA-TF for LWR Safety Analysis of Large Problems

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A matrix solver is implemented in COBRA-TF in order to improve the computation efficiency of both numerical solution methods existing in the code, the Gauss elimination and the Gauss-Seidel iterative technique. Both methods are used to solve the system of pressure linear equations and relay on the solution of large sparse matrices. The introduced solver accelerates the solution of these matrices in cases of large number of cells. The execution time is reduced in half as compared to the execution time without using matrix solver for the cases with large matrices. The

achieved improvement and the planned future work in this direction are important for performing efficient LWR safety analyses of large problems.

POLCA-T Simulation of OECD/NRC BWR Turbine Trip Benchmark Exercise 3 Best Estimate Scenario TT2 Test and Four Extreme Scenarios

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Westinghouse transient code POLCA-T brings together the system thermal-hydraulics plant models and the 3D neutron kinetics core model. Code validation plan includes the calculations of Peach Bottom end of cycle 2 turbine trip transients and low-flow stability tests. The paper describes the objectives, method, and results of analyses performed in the final phase of OECD/NRC Peach Bottom 2 Boiling Water Reactor Turbine Trip Benchmark. Brief overview of the code features, the method of simulation, the developed 3D core model and system input deck for Peach Bottom 2 are given.

The paper presents the results of benchmark exercise 3 best estimate scenario: coupled 3D core neutron kinetics with system thermal-hydraulics analyses. Performed sensitivity studies cover the SCRAM initiation, carry-under, and decay power. Obtained results including total power, steam dome, core exit, lower and upper plenum, main steam line and turbine inlet pressures showed good agreement with measured plant data. Thus the POLCA-T code capabilities for correct simulation of turbine trip transients were proved.

The performed calculations and obtained results for extreme cases demonstrate the POLCA-T code wide range capabilities to simulate transients when scram, steam bypass, and safety and relief valves are not activated. The code is able to handle such transients even when the reactor power and pressure reach values higher than 600 % of rated power, and 10.8 MPa.

Localized Void Feedback Effects under Single Rod Drop Transient in BWR

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A postulated single control rod drop transient was analyzed for a typical BWR plant. In order to include feedback effects of the intra-bundle void distribution in those bundles neighboring the withdrawn control blade, transient pin power distributions were reconstructed by the plant simulator TRAC/BF1-ENTRÉE and were exported to the subchannel code, NASCA.

The cross section libraries with flat and distorted void distributions were combined in the two-way coupling calculation. Exposure trends of neutronic properties were compared between two void distributions. Although the infinite multiplication factor wasn't notably influenced, the radial peaking factor changed significantly due to void distortions. A pin-by-pin concentration change of fissile materials also contributes to this result.

At the initial state, the void fraction in peripheral subchannels facing the withdrawn blade became prominently low. The power level in the surrounding bundles decreased due to larger neutron leakage caused by distorted void distributions. During transient, the linear power density at the critical fuel rod increased faster than that of the flat void model. Change in the fuel surface heat flux was attenuated due to the heat conduction time delay. The peak cladding temperature was slightly lower than that of the flat void model due to cancellation among these factors.

Peach Bottom-2 Low-Flow Stability Test using Trac-Bf1/Valkin and Relap5-Mod.3.3/Parcs Codes

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The possibility of instability in the core of a boiling water reactor induced by thermal-hydraulic and void reactivity feedback has been the subject of many analytical and experimental investigations. However, to improve the safety systems of these reactors, it is necessary to be able to detect in a reliable way these oscillations from the neutronic signals.

Four stability tests were developed in 1977 at end of cycle 2. These tests were conducted along the low-flow end of the rated power-flow line, and along the power-flow line corresponding to minimum recirculation pump speed. In this work, three dimensional time domain BWR stability analysis were performed on test point 3 (PT3) for the core wide oscillation mode.

We have carried out a number of perturbation analyses using two coupled codes: TRAC-BF1/VALKIN and RELAP5-MOD3.3/PARCS. A modal analysis of the 3D neutronic power evolution has also been calculated.

The purposes of these analyses are: to perform the real situation for comparing the results with real data and to compare the behaviour of the two coupled codes used.

For reducing the CPU time and looking for an unstable point, in most of the cases we have use a model with only one thermalhydraulic channel. The differences between the results obtained using 1 or 33 channels are not very important.

The calculated results show that point PT3 is a nearly stable point and it is at the end of cycle, while the obtained average axial power distribution shows a non bottom-peaked profile (stable). Nevertheless, the characteristics of the in-phase instability can be recognized in the different analyses

Bruce "B" Core Conversion.

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The CANDU reactor is one of the few commercial power reactors with a small positive void coefficient. Supplementary, a postulated large break loss of coolant accident (LLOCA) in the primary heat transport system will cause the relocation of fuel towards the inlet end of channels. With the current Bruce "B" Fuelling-against-Flow configuration, this will result in a moving of fuel bundles of low burnup to regions of greater neutron flux, introducing positive reactivity and increasing the power pulse. In order to improve LLOCA consequences and hence increase safety margins, Bruce "B" units are being converted to a Fuelling-with-Flow scheme. This will result in reversal of the fuel burnup profile relative to the coolant flow direction, eliminating the positive fuel string relocation reactivity.

This paper discusses the safety and implementation aspects of the core conversion that have been considered.

Coupled Neutronics-Thermalhydraulics Calculations for the Safety Analysis of the PBMR

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This paper presents an overview of the safety analysis of the pebble-bed reactor using the advanced coupled-code NEM-THERMIX. This coupled code is a combination of the Nodal Expansion Method code (NEM) and THERMIX-DIREKT. NEM is a three-dimensional (3-D) steady state and transient neutronics code developed at Penn State. The cylindrical geometry option of NEM has been verified on pebble bed modular reactor (PBMR) neutronics core benchmarks. THERMIX-DIREKT is a 2-D high temperature (HTR) thermal-hydraulics code extensively validated for PBMR analysis. The spatial and temporal coupling algorithms have been developed and the coupling has been tested demonstrating to be functional and verified through code-to-code comparisons. The coupled code is used in this work to illustrate the importance of coupled neutronics/thermal-hydraulics calculations when analyzing the safety of high temperature gas-cooled reactors. Using this code, it is demonstrated that the neutronics behavior is sensitive to cross section changes due to temperature and spectrum feedback effects. The spatial dependence of these effects is very important to be modeled and hence, coupled calculations are inevitable in establishing and building a safety case for the high temperature gas-cooled reactors of a pebble-bed type like the PBMR.

Session Organizers: Alain Santamarina (CEA Cadarache), Robert Jacqmin (CEA Cadarache), Richard Sanchez (CEA). Session Chair: Alain Santamarina (CEA Cadarache), Farzad Rahnema (Georgia Institute of Technology).

Experimental Validation of the APOLLO2 Code for High Burnup MOX Fuel. JEF2.2 Results and JEFF3.0 Improvements.

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This paper presents the APOLLO2 analysis of a destructive experiment of MOx fuels from the DAMPIERRE PWR reactor using the latest versions of the JEFF European nuclear data library. This analysis has demonstrated the ability of the current industrial package APOLLO2.5/CEA93.V6 to predict actinide inventory in high-burnup MOx fuel up to 60GWd/tHM.

This interpretation confirms the major JEF2.2 nuclear data trends given by the experiments performed in France and shows the improvement given by JEFF3.0 in the prediction of isotopic fuel contents, mainly U236, Np237 and Pu242. These MOx results are consistent with the satisfactory C/E comparison obtained with JEFF3.0 on UOx spent fuel inventory, except for Pu239 which remains overestimated: +4%±2% at 50GWd/tHM and +7%±3% at 60GWd/tHM.

Furthermore, it is expected that the next JEFF3.1 evaluation will improve the Pu239 C/E agreement as the U238 effective integral resonance should be decreased by 1%

Assessment of CASMO-4 Predictions of the Isotopic Inventory of High Burn-Up MOX Fuel

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This paper discusses the results of an assessment of the depletion model in CASMO-4 with highly-burnt MOX fuel data from the Programme ARIANE. Isotopic compositions calculated with CASMO-4 for fuel sample BM5 irradiated for 6 cycles (20-25) in Beznau-1 were compared with experimental data from PSI and SCK•CEN radio-laboratories. A 1-D Pin-Cell and a 2-D fuel assembly models were used, together with detailed and burn-up averaged irradiation histories. Neutronic data were provided by the ENDF/B-IV and JEF2.2 based CASMO-4 libraries. The results at the time of measurement showed that the inventory predictions for the most important actinides differed from the experimental values by less than 10%. They also highlighted large discrepancies in the calculation of Am and Eu content, and found inaccuracies, probably linked to the experimental techniques, with the estimation of ^{129}I and some metallic fission product inventories. Additional studies on the effect of refined intra-pin nodding showed an influence in the quality of the predictions of some actinides, eg Pu, Cm and Am, for which good self-shielding modeling is important.

PANDA Code Application to the OECD/NEA 2D/3D MOX Assembly Benchmark Calculations

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PANDA is an all purpose dicrete ordinates transport code developed at CEA/Bruyères-le-Châtel. It is applied to stochastic neutronics applications like backward time dependent neutron number and count fluctuations or extinction probability calculations. The code can also be used for more conventional applications like the effective multiplication factor. It is based on a classic discrete ordinate (S_N) transport solver which can be used for 1D, 2D and 3D computations. In order to validate PANDA code in two and three spatial dimensions, benchmark computations are performed.

In this paper we present PANDA results on the international OECD/NEA C5G7 MOX benchmark. This test problem was designed by the expert group on 3D radiation transport benchmark of the OECD/Nuclear Agency in order to assess the ability of current deterministic transport codes to treat two and three dimensional core problems without homogenisation.

The obtained solutions seem to be not fully converged with regard to the MCNP reference solutions. Accurate solutions in 3D will require a high level of spatial and angular discretization leading to very expensive calculations in term of CPU time and memory usage.

Benchmark Analysis of the DeCART MOC Code with the VENUS-2 Critical Experiment

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Computational benchmarks based on well-defined problems with a complete set of input and a unique solution are often used as a means of verifying the reliability of numerical solutions. VENUS is a widely used MOX benchmark problem for the validation of numerical methods and nuclear data set. In this paper, the results of benchmarking the DeCART (Deterministic Core Analysis based on Ray Tracing) integral transport code is reported using the OECD/NEA VENUS-2 MOX benchmark problem. Both 2-D and 3-D DeCART calculations were performed and comparisons are reported with measured data, as well as with the results of other benchmark participants. In general the DeCART results agree well with both the experimental data as well as those of other participants.

Monte Carlo Analysis of High Moderation 100% MOX BWR Cores using JEF2 and JENDL3 Nuclear Data

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This paper presents a Monte Carlo analysis of the BASALA neutron physics programme implemented in the EOLE facility at CEA Cadarache. This programme comprised two types of cores simulating hot (BASALA-H) and cold (BASALA-C) conditions. Reference and “poisoned” configurations were investigated in both cases.

The BASALA-H core included a configuration with extra water rods in order to simulate overmoderated conditions and a configuration with fuel pin thick overclads in order to simulate 70% void conditions while the BASALA-C core included configurations with crossshaped control blades. Different kinds of physical parameters were measured and calculated as the effective multiplication factor, reactivity worth of control blades, gadolinium pins and fission rate distributions. The experiment and calculation comparisons are satisfactory and demonstrate the ability of French and Japanese reference calculation tools to predict the main neutronic parameters of 100% MOX BWRs.

6C. Physics Code Validation

Session Organizers: Alain Santamarina (CEA Cadarache), Robert Jacqmin (CEA Cadarache), Richard Sanchez (CEA). Session Chair: Alain Santamarina (CEA Cadarache), Farzad Rahnema (Georgia Institute of Technology).

Experimental Validation of Pin Power Distributions for a BWR Assembly with Hafnium Control Blades

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The accurate estimation of reactor physics parameters related to the presence of cruciform absorber blades in BWRs is important for safety assessment and for achieving a flexible operation during the cycle. Characteristics that are affected strongly include the pin power distribution for controlled regions.

The Paul Scherrer Institute and the Swiss Nuclear Utilities are conducting experimental reactor physics investigations and code validation activities related to modern LWR fuel as employed in the Swiss nuclear power plants, the so-called LWR-PROTEUS programme. A part of LWR-PROTEUS Phase I was devoted to the characterization of highly heterogeneous BWR fuel assemblies in the presence of absorber blades. The current paper presents the comparison of controlled-assembly pin power distributions calculated (C) by CASMO-4, HELIOS and BOXER with experimental (E) values measured in a Westinghouse SVEA-96+ assembly under full-density water moderation conditions in the presence of Westinghouse hafnium absorber blades. The results using detailed models show that typical *rms* deviations of the pin-by-pin (C-E) distributions are in the order of 1.5-1.6% for the three codes, a remarkably good agreement which is in fact quite similar to that obtained for the non-bladed configurations investigated in LWR-PROTEUS Phase I.

Validation of Integrated Burnup Code System SWAT2 by the Analyses of Isotopic Composition of Spent Nuclear Fuel

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This paper provides validation results of SWAT2, the revised version of SWAT, which is a code system combining point burnup code ORIGEN2 and continuous energy Monte Carlo code MVP, by the analysis of post irradiation examinations (PIEs). Some isotopes show differences of calculation results between SWAT and SWAT2. However, generally, the differences are smaller than the error of PIE analysis that was reported in previous SWAT validation activity, and improved results are obtained for several important fission product nuclides. This study also includes comparison between an assembly and a single pin cell geometry models.

Verification of Lattice Analysis Method Through BWR UO2 PIE Data Analysis

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Calculations with a lattice analysis code have been done for the experimental data of the rod-by-rod distributions of FP nuclides in the BWR fuel assemblies with the burn up form 12.6 to 48.8 GWd/t and for the pellet average isotope compositions of actinide nuclides (U, Pu, Am and Cm) and major FP nuclides with the pellet burn up from 16.7 to 44.0

GWd/t. The experimental data are those for the 8x8 UO2 fuel assemblies that were discharged from the Unit 2 of Fukushima power station 2 (2F2).

The SRAC cord system coupled with the JENDL-3.2 nuclear data library has been used for the burn-up calculations. For the rod-by-rod distributions of the FP nuclide, the values of C-E (%) are from 0 to 4 % except for the Gd2O3-UO2 fuel rod. The root mean square of the difference between the analysis results and the measurements of the 7-rod positions is almost in two times of the statistical errors of the measurements. For the pellet average isotope compositions of actinide nuclides, the comparison between the analysis and measurement results has been done and discussed in terms of the measurement errors and analysis methods.

Uncertainty Analysis Applied to Fuel Depletion Calculations

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A newly developed methodology is presented, based on a statistical approach, for assessing uncertainties in lattice code calculations due to uncertainties in fuel description; viz. depletion conditions, initial isotopic content and fuel assembly dimensions. Its application can quantify the uncertainties in the estimated nuclide compositions at any time during the irradiation history and compare them with the experimental uncertainties. It has been applied a CASMO-4 depletion analysis of a well characterized MOX sample, BM5, from the ARIANE Programme. Estimates of uncertainties in nuclide concentration predictions were obtained through the statistical processing of 100 CASMO-4 calculations, which yielded *tolerance intervals* of 95% probability content and 95% confidence. The results have shown that important information on the quality of the code's estimates can be obtained by comparing them and their associated uncertainty to experimental data and their reported errors. The methodology also permits to obtain corresponding burn-up dependent uncertainty estimates for neutronic parameters such as k_{inf}, Σ_a, $\nu\Sigma_f$, etc.

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Won Sik Yang (ANL), Thomas J. Downar (Purdue University).

Neutronics Codes Currently Used in Japan for Fast and Thermal Reactor Applications

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Neutronics codes currently used in Japan for core design and core management calculations are reviewed, and the features of the codes are described together with description of reactor core improvement. The review covers pressurized water reactors, boiling water reactors and fast reactors.

HORUS3D Code Package Development and Validation for the JHR Modeling

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The definition and the validation of the HORUS3D (Horowitz Reactor simulation Unified System) package, described in this paper, is dedicated to the design studies of the future European MTR, called Jules Horowitz Reactor.

HORUS3D/N, based on APOLLO2 and CRONOS2 codes, allows the neutronics modeling of the Horowitz reactor with great confidence compared to MC reference calculations. An experimental program, organized in two phases, is planned. The first phase, called VALMONT, will allow the qualification of the fuel reactivity calculation. The second phase, named AMMON, aims the qualification of the safety parameter calculation.

HORUS3D/P is devoted to photonics calculations, taking into account the actual assembly geometry. An experimental program, ADAPh, is planned in order to diminish the important nuclear data uncertainties.

HORUS3D/Cy, based on the DARWIN package, allows to determine cycle design parameters such as residual heat with great confidence. The experimental program IRIS-γ will allow to extend the present qualification domain to the UMoAl fuel depletion.

HORUS3D/Th and HORUS3D/Sys, the Core & System thermal-hydraulics parts, are based on FLICA4 and CATHARE2 codes. The specific characteristics of the JHR element led to a specific experimental program on the SULTAN-JHR test facility.

At present, the first version of the HORUS3D package is being used for the Detailed Design Studies of the Horowitz Reactor.

On Some Features of Quasi-Static Schemes in Reactor Kinetics

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The quasi-static method has shown to be an efficient tool for the time-dependent analysis of nuclear systems allowing a significant reduction of the computational effort needed for full space and energy transient calculations. The method is based on the factorization of the neutron density in the product of a shape and an amplitude function. The time integration of the balance equations is then carried out on a fast scale for the amplitude and on a much slower scale for the shape. Since the shape calculation is the most time consuming, this scheme allows a significant reduction of the computing time. In this work some specific features of the quasi-static technique are investigated, with special regard to the application of the method to source-driven systems. In particular, the effect of different choices of the weighting

function are investigated. Paradigmatic configurations are considered in diffusion theory, thus allowing to calculate analytically the full time behavior of the neutron flux. Then the transient for the exact amplitude and shape functions is derived for various choices of the weight, having imposed the normalization condition. These functions are used to compute the kinetic parameters at the beginning of the transient and perform the corresponding point kinetic calculation. The scheme can be included into a quasi-static procedure, updating the kinetic parameters during the transient using the exact shape.

The results presented show that, while for systems evolving from a nearly-critical situation the most obvious choice of the weighting function to generate the kinetic parameters to be used during a shape step is the reference system importance function, this may be questionable when passing to source-driven systems. It is possible to evidence transients where the most effective weighting function turns out to be either a source-driven adjoint or even simply a constant function. The study of a scheme to enhance the effectiveness of quasi-statics shows that there are wide possibilities to improve the performance of the procedure, still retaining the accuracy of the results.

Albedo Conditions for Multigroup Anisotropic Scattering Models of Nuclear Reactors

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We report some improvements in the theory of spectral nodal methods for the solution of nuclear reactor problems in the discrete ordinates (S_N) formulation of neutron transport theory. Specifically, we present exact and numerically stable multigroup S_N albedo conditions that substitute the axially oriented, geometrically thick reflector regions of thermal reactors in S_N eigenvalue calculations with arbitrary (Legendre) order of anisotropic scattering. These conditions are expected to improve the accuracy and to increase the efficiency of *any* slab-geometry S_N code grounded in spatial discretization methods that are approximate and in the transport technique of sweeping the space-angle grid with the discretized S_N equations. These overall improvements can be achieved by removing the reflector regions from the transport sweeps and by replacing them with our multigroup S_N albedo conditions. The relevance of our multigroup S_N albedo conditions is indicated by means of numerical experiments with an axial model for a thermal reactor core and a diamond difference code. In addition, the conditions derived here may serve as a basis for improvements in approximate S_N albedo conditions recently developed for x,y-geometry nuclear reactor calculations, as these approximate conditions are derived for one-speed models with isotropic scattering and, to our knowledge, they are not numerically stable.

An Algebraic Multigrid Resolution Strategy for the DP_N Synthetic Acceleration Operator

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In some recent papers, the DP_N operator has been introduced as a mean to accelerate free iterations for TDT code that is based on the Method of Characteristics (MOC). The quite complicated structure of this sparse matrix operator asks for an adapted numerical treatment. In the past many improvements of classical numerical algorithms, i.e. Krylov subspaces methods, have been proposed: these improvements consist basically in an adaptation of the standard algorithms to the specific form of DP_N operator. In this paper we conclude this task by presenting the impact of a newly implemented technique of Algebraic Multi-Grid. We present here an AMG method tailored for our specific application. We show that, by means of a preliminary elimination phase, we can obtain a reduced dimension operator, with exactly the same form of the full dimension one. The solution is then obtained by solving this reduced problem with the same techniques that have been previously developed. Results on the C5G7 benchmark are shown that prove the method to be efficient and fast when applied to realistic assembly calculations.

Convergence Analysis of 2-D/1-D Coupling Methods for the Three-Dimensional Neutron Diffusion Equation

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A convergence analysis was performed for three methods used to couple the 2-D radial and 1-D axial solutions of the 3-D neutron diffusion equation. In the first and second methods, the axial net currents and partial currents at plane interfaces were used for the inter-plane couplings, respectively. In a newly proposed third method, the current correction factors from the axial two-node kernel are used to couple the planes similar to the conventional CMFD formulation with a 2-node kernel. The new method has at least two advantages compared to the other methods. First, it is always stable whereas the net current method diverges for small mesh sizes. Second, the new method uses a Gauss-Seidel planar sweeping, and in the range of practical mesh sizes provides the best performance in terms of convergence rate.

Numerical Convergence Analysis of the CMFD Method for the Three-Dimensional Two-Group Diffusion Problem

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The convergence of the CMFD method was examined for three-dimensional, two-group diffusion problems. Two test problems were investigated: a homogeneous 3-D 2-G model eigenvalue problem and the NEACRP LWR transient benchmark problem. A One-Node (1-N) kernel was implemented into PARCS for both steady state and transient problems which complemented the existing Two-Node (2-N) kernel. The convergence rates of both the one and two node kernels were measured numerically with emphasis on the convergence of current correction factors (CCFs). This paper presents the numerical test results and analyzes them in the context of the Fourier analysis with simple model problems presented in previous publications by the authors.

An Approach of Super-Element Sweeping for the Solution of Neutron Transport Equation in Heterogeneous Geometry

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We describe a new solution method for the discrete ordinates equations based on super-element sweeping for 2-D and 3-D neutron transport calculations. A super-element is a unit of sweeping calculation and may consist of several heterogeneous meshes.

Unlike existing S_N sweeping methods, super-element sweeping method performs sweeping calculation with point value angular fluxes of a super-element. This super-element sweeping method can treat heterogeneous geometries and it costs less computation time compared with other transport methods for heterogeneous geometry such as the method of characteristics and collision probability method.

The paper provides the results of the method's applications to homogeneous and heterogeneous problems.

Efficient Hybrid NEM/BEM Transient Method

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Among other transport methods the multi-group pin-by-pin Simplified P_3 (SP_3) approximation is found to be the most applicable choice fro implementation into the preexisting core simulator code. This paper describes the implementation of the multi-group pin-by-pin SP_3 method in the Pennsylvania State University's (PSU) NEM core transient simulator. First, A_2 Boundary Element Method (BEM) is implemented in three-dimensional Cartesian geometry for local solution on pin-by-pin basis. An efficient scheme was introduced to couple local (A_2-BEM) solution with global core (NEM Response Matrix – RM) calculation. Second, direct nodal SP_3 pin-by-pin method was implemented also in the framework of the NEM code.

Case 3 of the international OECD Benchmark L-336 [1] has been used as a test problem. The two new options, implemented in NEM–SP_3 nodal pin-by-pin and NEM-RM/A_2-BEM, demonstrated to be functional and performed as expected. The hybrid NEM-RM/A_2-BEM model with eight energy groups has also been applied to the VENUS II international benchmark, where actual measured data is available [2]. The deviation in the predicted k_{eff} as compared to experimental value of 1.00000 was 250 pcm and the deviations in the prediction of pin powers as compared to reference values were in the range of ± 2.5 %.

7A. Non-Conventional Reactors

Session Organizer: Jan Leen Kloosterman (Delft University of Technology).
Session Chair: Danny Lathouwers (Delft University of Technology).

Simulation of Caliban Reactor Burst Wait Time and Initiation Probability Using a Point Reactor Model and PANDA Code

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Caliban fast metal reactor is designed for pulsed neutron flux generation when operated above prompt critical. When these pulsed experiments are performed without an external neutron source there is a non reproducible time delay between the prompt critical state and the neutron burst. This wait time duration can be up to several seconds. A similar phenomena was observed with GODIVA II reactor and theoretically explained by G. E. Hansen. The wait time fluctuation is a stochastic neutron transport problem, it results mainly from the probability for one source neutron to initiate a persistent fission chain.

The experimental wait time probability distribution is obtained using time measurements performed during burst experiments with Caliban. A stochastic neutronics time dependent with delayed neutrons point model developed by B. Lemaire is used to calculate the wait time probability distribution. The numerical results given by this model are in good agreement with the experiments.

The initiation probability with position and velocity dependencies is solution of the non linear stationary adjoint transport equation which is solved using PANDA discrete ordinates code.

Passive Decay Heat Removal in a Fluidized Bed Nuclear Reactor

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Numerical simulations have been performed to investigate the possibility of passive decay heat removal in a fluidized bed nuclear reactor. The conduction, convection and radiative heat transfer were included and the high Reynolds number k-ε turbulence model was applied for the flow calculations. A P_1 approximation method for participating media was used for the radiation model. The reactor was modeled as a 2D axisymmetric cylinder. Calculations were performed for different initial total operating power to see whether the associated decay heat can be removed by passive means without compromising the safety margin. The results show the decay heat can be removed by passive means up to operating powers of about 55 MW. Further the results showed that all heat transfer mechanisms are of importance to remove heat from the bed of fuel particles to the reflector.

Molten Salt Reactors and Possible Scenarios for Future Nuclear Power Deployment

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An important fraction of the future energy demand may be satisfied by nuclear power. In this context, the possibilities of worldwide nuclear deployment are studied. We are convinced that the Molten Salt Reactors may play a central role in this deployment.

The Molten Salt Reactor needs to be coupled to a reprocessing unit in order to extract the Fission Products which poison the core. The efficiency of this reprocessing has a crucial influence on reactor behavior especially for the breeding ratio. The Molten Salt Breeder Reactor project was based on an intensive reprocessing for high breeding purposes. A new concept of Thorium Molten Salt Reactor is presented here.

Including this new concept in the worldwide nuclear deployment, to satisfy these power needs, we consider three typical scenarios, based on three reactor types: Pressurized Water Reactor, Fast Neutron Reactor and Thorium Molten Salt Reactor.

The aim of this paper is to demonstrate, in a first hand that a Thorium Molten Salt Reactor can be realistic, with correct temperature coefficients and at least iso-breeder with slow reprocessing and new geometry; on the other hand that such Molten Salt Reactors enable a successful nuclear deployment, while minimizing fuel and waste management problems.

Closing the PWR Fuel Cycle with a Molten-Salt Incinerator

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We present a detailed study of a non-moderated Molten-Salt (MS) reactor, when operated as a transuranium elements burner, closing the nuclear fuel cycle of a PWR reactor. We evaluate its neutronic characteristics and main safety parameters, both in the first years of operation and in equilibrium fuel cycle. Influence of the carrier salt composition, neutron flux and feed composition on the burner performance are also studied.

Our results show, that non-moderated MS burner can be operated as a critical burner reactor without fertile support in an acceptable range of parameters. The temperature feedback coefficients are relatively low, but we show that their further optimization is possible. Delayed neutron fraction attains a similar value as in the case of fast reactors. Neutron balance allows to feed the burner with a PWR fuel having a very high burn-up. This gives a margin for further optimization of PWR fuel cycle, hopefully leading to the lowering of the proportion of dedicated reactors needed to close the PWR fuel cycle.

Monte Carlo Calculation of the Effects of Delayed Neutron Precursor Transport in Molten Salt Reactors

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Due to the motion of the fuel in molten salt reactors a portion of the delayed neutrons is born outside the reactor core, which decreases reactivity. The authors in their earlier papers presented a method which was developed in order to modify the program MCNP so that it can take into account the transport of the delayed neutron precursors, and therefore capable of calculating the reactivity loss of reactors with circulating fuel.

In the present paper, a further development of the method and its implementation in the MCNP code is presented, which makes it possible to radically reduce (by four orders of magnitude) the computer time needed to calculate the reactivity loss at the same accuracy. The method is based on the fact that the above mentioned decrease in reactivity is influenced by the delayed neutrons only, and can be outlined as follows. In the first of two consecutive criticality (KCODE) calculations the data describing the fission source (such as spatial coordinates of the fission points, delayed neutron parameters etc.) are recorded in a file, while in the second run only the random walk of the delayed neutrons is simulated using the data read from the file.

Studies of Physical Features of Cascade Subcritical Molten Salt Reactor with External Neutron Source

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Paper presents the results of computational analysis of cascade schemes of a molten salt subcritical reactor (CSMSR) – burner of long-lived radioactive wastes with the feed isotopic composition formed in the closed fuel cycle of the multicomponent nuclear power system. In the considered equilibrium model of the nuclear power with the closed fuel cycle, minimization of quantities of minor actinides serves as the optimization criterion in determination of shares of installed capacity of reactor of different types.

In this paper we consider the CSMSR (with the proton accelerator and target as a neutron source) as an element of the multicomponent nuclear power system, which is responsible for minor actinides utilization.

In the present moment one can hardly propose a technical solution for reaching the high level of neutron flux in the transmutation zone in a subcritical reactor with subcriticality in the range of 1-5% K_{eff} (K_{eff} is in the range of 0.99-0.95).

The paper presents studies of the possibility to use special intermediate zone with high importance of fast neutrons that allow to achieve neutron flux level, sufficient for effective transmutation under condition of fixed accelerator power. Several CSMSR reactor designs are considered with molten salt and solid fuel zones of cascade amplification.

7B. Nuclear Data

Session Organizers: Luiz Leal (ORNL), Mark Chadwick (LANL).
Session Chairs: Russell Mosteller (LANL), Alain Santamarina (CEA Cadarache).

New Neutron Cross Section Measurements at ORELA for Improved Nuclear Data

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The Oak Ridge Electron Linear Accelerator (ORELA) was used to measure neutron total and capture cross sections of aluminum, silicon, chlorine, fluorine, and potassium in the energy range from 100 eV to ~600 keV. These measurements were carried out to support the Nuclear Criticality Safety Program (NCSP). Concerns about the use of existing cross section data in nuclear criticality calculations have been a prime motivator for the new cross section measurements. Our results are substantially different from the evaluated nuclear data files of ENDF/B-VI and JENDL-3.2.

Measurement and Calculation of the ^{233}Pa Fission Cross-Section for Advanced Fuel Cycles

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The energy dependence of the neutron-induced fission cross-section of ^{233}Pa has been measured directly for the first time from the fission threshold up to 8.5 MeV. This fission cross-section is a key ingredient in feasibility studies on fast reactors and accelerator driven systems based on the Th-U fuel cycle. The results are at strong variance with the existing evaluations. The new experimental data give lower cross-section values and resolve the question about the threshold energy. Additionally a new theoretical calculation of the reaction cross-section has been performed with the statistical model code STATIS, showing a very good agreement with the experimental data.

Scattering Law Data for Graphite in Gas Cooled High Temperature Reactors

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For polycrystalline graphite the presently available scattering law data were re-evaluated to prove their actuality for application of gas cooled reactor studies. Several frequency distributions $\rho(\omega)$ were considered published by Hawari et. al. (NCSU), Nicklow et al. (recently processed by Difilippo et al., ORNL) and Young, Koppel (GA). To get numerical accurate results care was taken to have a correlation of the alpha and beta grid to the maxima and minima of $\rho(\omega)$. The generated scattering law data were used as basis for the calculation of differential and integral neutron cross sections which could be compared with corresponding experimental data. For all investigated models the results agree well with measurements but the overall agreement of the Young, Koppel model (GA) seems to be slightly better than the NCSU and the ORNL model. Especially, the GA model reproduces well the measured cross sections and specific heat data.

The generated data sets were then used to calculate thermal neutron spectra and integral data for different configurations with graphite as moderator. The measured neutron flux spectra in poisoned graphite were sufficiently reproduced by all models. In pure graphite there are some differences in the maximum, the ORNL data result in a slightly harder spectrum compared to IKE/GA and NCSU data and experimental values. However, integral parameters such as k_{eff} for UOX, MOX and ThOX fuel show no significant sensitivity of the regarded scattering law models.

Session Organizers: Luiz Leal (ORNL), Mark Chadwick (LANL).
Session Chairs: Russell Mosteller (LANL), Alain Santamarina (CEA Cadarache).

An Unresolved Resonance Evaluation for 235U

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This paper discusses the resonance evaluation of the 235U cross sections in the unresolved resonance region from 2.25 keV up to 25 keV and its use in benchmark calculations for criticality safety applications.

A criticality safety calculation for a nuclear system with an energy spectrum that peaks in the intermediate energy region requires accurate neutron cross sections in both the resolved and the unresolved resonance regions. A resolved resonance region evaluation of the 235U cross section was performed in the 90's, and the evaluation has greatly improved results of benchmark calculations. Average values of the Reich-Moore resonance parameters obtained in the resolved region were used to initiate a new resonance evaluation of the 235U cross sections in the unresolved resonance region. The experimental data used in the analysis consisted of high-resolution transmission data, plus fission and capture cross-section data measured at the Oak Ridge Electron Linear Accelerator (ORELA). The evaluation was performed using the computer code SAMMY, which incorporates a methodology for data evaluation in the unresolved resonance region.

Status of a New Evaluation of the Neutron Resonance Parameters of ^{238}U at ORNL

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Neutron resonance parameters were obtained in the energy range 0 keV to 20 keV from a sequential SAMMY analysis of high resolution neutron transmissions and capture cross sections measured at the Oak Ridge Electron Linear Accelerator (ORELA). In the energy range 0 keV to 10 keV the analysis used as prior values the resonance parameters of ENDF/B-VI evaluated file. In the energy range10 keV to 20 keV, where ENDF/B-VI uses average parameters for the description of the cross sections, this analysis resulted in the creation of a set of resolved resonance parameters which will allow more accurate calculation of the self shielding factors. The statistical properties of the parameters in the energy range 0 keV to 20 keV are given. The results of selected benchmark calculations show significant improvements compared to calculations using the ENDF/B-VI evaluation, particularly for the k_{eff} prediction in thermal benchmarks.

JEF2.2 Nuclear Data Statistical Adjustment Using Post-Irradiated Experiments

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This paper presents a statistical adjustment study on JEF2.2 nuclear data library using several types of integral measurements. A new code called RDN based on non-linear regression methods using iterative technique was developed for this purpose. Relevant measurements performed in various critical facilities, mostly in thermal and intermediate neutron spectra, were selected for this study and a special attention was paid to the inclusion in the adjustment of the results from spent fuel experiments (measured actinide concentrations as a function of burn-up in PWR UO_2 and MOX assemblies).

Valuable results have been obtained for thermal and resolved range data, which confirmed the overall quality of JEF2.2 cross-sections and multiplicities for thermal reactors. These results support the new evaluations of the resolved resonances for U235 and Pu241 in JEFF3.0, and suggest new trends for U238, Pu239 and Pu240 cross sections.

Session Organizers: Kostadin N. Ivanov (Pennsylvania State University), Werner Maschek (Karlsruhe).
Session Chairs: Dobromir Panayotov (Westinghouse Atom AB), Tim Newton (Serco Assurance).

OECD/DOE/CEA VVER-1000 Coolant Transient (V1000CT) Benchmark for Assessing Coupled Neutronics/Thermal-Hydraulics System Codes for VVER-1000 RIA Analysis

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The present paper describes the two phases of the OECD/DOE/CEA VVER-1000 coolant transient benchmark labeled as V1000CT. This benchmark is based on a data from the Bulgarian Kozloduy NPP Unit 6. The first phase of the benchmark was designed for the purpose of assessing neutron kinetics and thermal-hydraulic modeling for a VVER-1000 reactor, and specifically for their use in analyzing reactivity transients in a VVER-1000 reactor. Most of the results of Phase 1 will be compared against experimental data and the rest of the results will be used for code-to-code comparison. The second phase of the benchmark is planned for evaluation and improvement of the mixing computational models. Code-to-code and code-to-data comparisons will be done based on data of a mixing experiment conducted at Kozloduy-6. Main steam line break will be also analyzed in the second phase of the V1000CT benchmark. The results from it will be used for code-to-code comparison.

The benchmark team has been involved in analyzing different aspects and performing sensitivity studies of the different benchmark exercises. The paper presents a comparison of selected results, obtained with two different system thermal-hydraulics codes, with the plant data for the Exercise 1 of Phase 1 of the benchmark as well as some results for Exercises 2 and 3.

Overall, this benchmark has been well accepted internationally, with many organizations representing 11 countries participating in the first phase of the benchmark.

Uncertainty and Sensitivity Analysis Applied to Coupled Code Calculations for a VVER Plant Transient

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The development of coupled codes, combining thermal-hydraulic system codes and 3D neutron kinetics, is an important step to perform best-estimate plant transient calculations for safety analysis. It is generally agreed that the application of best-estimate methods should be supplemented by an uncertainty and sensitivity analysis to quantify the safety margins. Coupled codes have been developed for all reactor types like PWRs, BWRs and also Russian VVERs. The paper refers to several EU activities that were focused on the validation of coupled codes for VVER. Within the frame of the VALCO project of the 5th EU research programme the GRS uncertainty and sensitivity method was applied for a VVER-440 plant transient, which was already studied earlier for validation of coupled codes. The main steps of the GRS uncertainty and sensitivity method will be described. The GRS method is generally applicable and represents a quantitative method based on proven statistical methods and is supported by the statistical code package SUSA. Typical results from the analysis of the plant transient by several working groups using coupled codes are described. The results demonstrate the capability of an uncertainty and sensitivity analysis for plant transients.

Session Organizers: Kostadin N. Ivanov (Pennsylvania State University), Werner Maschek (Karlsruhe).
Session Chairs: Dobromir Panayotov (Westinghouse Atom AB), Tim Newton (Serco Assurance).

Study of the Nuclear Fuel Behavior with Coupled 3D Neutronics/Thermal-Hydraulic Codes

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This paper analyzes the behavior of the nuclear fuel of a WWER-1000 reactor during a Rod Ejection Accident (REA). The aim of the work was to establish also a complete methodology for execute calculations in the future on high burnup fuels using the state-of-the-art of the nuclear codes. The tools used for the analysis were a 3D neutronics / thermal-hydraulics coupled codes and a 2D neutronics transport code for the cross section libraries generation. The study performed several calculations analyzing the effect of different cross section modelling and of different status of the plant on the evolution of the accident. Sensitivity analyses were also carried on modifying various physical parameters of the nuclear fuel. The results show that the Hot Zero Power transient, with the fuel at the End of Cycle and with the plant having only two pumps in operation, is the more dangerous situation for the REA. Massive ebullition of the channels and DNB on the clad surfaces were discovered. The Energy Release to the Fuel calculated is below the safety limits, but some concerns remain, especially thinking to a fuel with an increased burn-up.

TRAC-M/AAA Code Assessment for Transient Analysis of Pb-Bi Cooled Fast-Spectrum Reactor Systems

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A coupled system of codes is currently under development at the Paul Scherrer Institute (PSI) for the comprehensive transient analysis of fast-spectrum critical and sub-critical reactors cooled by liquid metal or gas. The thermal-hydraulic calculations in this code system will be performed by the TRAC-M/AAA code version, specially developed to simulate liquid-metal and gas coolants. The purpose of the presented work is to assess some of the models employed in this code, which are important for transient analysis of fast-spectrum reactor systems cooled by heavy liquid metals. In particular, the code predictions have been verified against (a) three sets of experimental data on two-phase heavy metal/gas flow and (b) the predictions of other codes in the framework of a beam-trip calculational benchmark for a Pb-Bi cooled ADS. The main conclusion of the work is that the TRAC-M/AAA code can, with reasonable accuracy, predict the considered phenomena in fast-spectrum reactor systems cooled by heavy liquid metals.

Source and Reactivity Perturbations in Accelerator Driven Systems with Conventional MOX and Advanced Fertile Free Fuels

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Comparative safety analyses and investigations have been performed for a small scale ADS with conventional MOX fuel and an 800 MWth power class ADT with advanced fertile free fuel, both cooled by Pb/Bi. The analyses cover perturbations of the source, as e.g. unprotected transient over current (UTOC) and beam interruptions as well as perturbations on the core side, protected/unprotected transient over power (P/UTOP), induced by reactivity additions and unprotected loss of flow (ULOF) accidents. It shows that the small scale ADS has a very good safety performance, while for the 800 MWth ADT with ZrO_2 matrix based fuel some safety problems are identified, mainly related to the large positive void feedback. Further design and safety optimizations are under consideration.

Safety Characteristics of Candidate Oxide Fuels for Accelerator Driven Transmuters

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The FUTURE Programme (Fuels for the Transmutation of Trans-Uranium Elements), established within the 5th European Framework Programme on Partitioning and Transmutation, has the main objective to design dedicated fuels for an Accelerator Driven Transmuter (ADT) based on well known oxide fuel technology. Dedicated oxide fuels intended for ADT systems, consisting of innovative oxide compounds in a solid solution which is diluted in an inert matrix, have not so far been developed in Europe or in the international scientific community. The principal innovation is related to the high minor actinide content and the absence of a fertile component. This paper presents the work that has been carried out to assess the basic feasibility of ADT cores utilising the candidate fuels selected within the FUTURE programme. Two particular aspects have been addressed in detail, the impact and behaviour of the different inert matrix materials, and an evaluation of their safety characteristics as a function of fuel irradiation and fuel isotopic content. Results for core performance and safety parameters are presented and discussed. These studies have been performed by Serco Assurance within the context of the European FUTURE programme and are sponsored by BNFL.

Beam Calculation for TRIGA Reactor

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As part of the INIE project of the Big-10 consortium, efforts are underway at UIUC to assemble and test a set of codes that can be used for the design of the next generation of research reactor. SCALE 4.4 has been selected as one of the set of codes for this purpose. We here report some results of analysis of a TRIGA reactor using KENOVI, one of the codes in the SCALE system. Results are presented for a beam optimization study. Effects of several parameters on the level and spectrum of flux are reported.

Thermal Neutron Time-of-Flight Spectroscopy at Penn State using a Single-Disk Chopper

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A single-disk, "slow" chopper system has been developed at the Penn State Radiation Science and Engineering Center (RSEC) for the purpose of energy spectrum measurements on thermal neutron beams. This was achieved by gating the beam with a single rotating narrowly-slotted disk of neutron-absorbing material (a "chopper") and measuring the resultant time-of-flight (TOF) distribution of transmitted neutrons to an adjacent detector, located at a known separation distance. The TOF distribution was transformed to distributions in neutron speed and energy, which closely matched models based on the Maxwell-Boltzmann distribution. The advantage of this technique is its simplicity; no measurement of the neutron or secondary particle energy deposition is necessary.

UIUC's Contribution to Big-10's INIE Project

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Recent progress toward assembling and testing a set of codes for the design of a research reactor is reported. Graphical user interfaces are developed for ORIGEN and DIF3D. KENOIV is used to optimize the beam design of a TRIGA reactor (reported in a companion paper). Results of recent development of a virtual environment for a radiation lab as well as for a URTR control room are also reported.

Modeling of Existing Beam-Port Facility at PSU Breazeale Reactor by using MCNP5

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The Radiation Science and Engineering Center facilities at the Pennsylvania State University include the Penn State Breazeale Reactor, gamma irradiation facilities, and various radiation detection and measurement laboratories. Due to inherited design issues with the current arrangement of beam ports and reactor core-moderator assembly, the development of innovative experimental facilities utilizing neutron beams is extremely limited. Therefore, a new core-moderator location in PSBR pool and beam port geometry was needed to be developed. A study is underway with the support of DOE-INIE funds to examine the existing beam ports for neutron output and to investigate new moderator and beam-port designs to produce more useful neutron beams.

The overall system for this study consists of two major parts, the core model and beam port model. Core calculations are performed by using a three dimensional nodal diffusion code ADMARC-H. Beam port calculations are performed with the MCNP code. An interface program has been developed at PSU to link the diffusion code to the neutron transport code. The MCNP model consists of the D_2O tank, graphite reflector block, and beam port tube with their surroundings. The results of the PSU package show good agreement with the experimental data.

AGENT Code: Open-Architecture Analysis and Configuration of Research Reactors – *Neutron Transport Modeling with Numerical Examples*

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The current state of development of codes that address the flexible neutron transport modeling in research reactor core geometries is extremely limited. The common method for accurate modeling of particle interactions is a Monte Carlo technique. While it provides great geometrical flexibility and high calculational accuracy, it suffers from long computational times, cumbersome modeling requirements, tedious and error-prone process of generating input and knowledgeable evaluation of numerically produced statistical errors. Highly accurate modeling of neutron trajectories within heterogeneous systems and within reasonable computation runtimes is possible using the MOC. The neutron random walks of Monte Carlo are replaced with a large number of parallel paths in a number of discrete directions in MOC to calculate flux and source data. We combine the MOC with the powerful mathematical tool of R-function theory to achieve the flexibility and accuracy of Monte Carlo codes, yet retain the speed and simplicity of deterministic codes. The AGENT code system will comprise a "*virtual reactor*" where users can configure core fuel arrangements, reactor experiments, and advanced reactor configurations. This will permit the community of URTR operators and experimenters to run a virtual reactor and monitor reactor behavior and performance. It will also support distance educational and outreach programs where remote access to the code suite will allow educators to perform virtual reactor demonstrations. The AGENT code is optimized for robustness, simplicity, accuracy, and efficiency. It supports a full treatment of neutron transport in highly heterogeneous geometries such as research reactors. The robustness of the geometrical module is assured through the sequential generation of the layers of geometry and automatic submeshing based on the R-function solid modeler. The simplicity of geometry description and selection of parameters for accurate treatment of neutron propagation is achieved through the hierarchical organization of simple primitives into complex domains. The accuracy is comparable to Monte Carlo codes and is obtained by following neutron propagation through real geometrical domains. The efficiency is maintained through a set of acceleration techniques introduced in all important calculation levels.

A Three Dimensional Two Energy Group Coupled Reactor Physics and Thermal Hydraulics Code (M32) – A Tool for Student Design Studies

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This paper describes a new code M32 that has been developed to perform coupled reactor physics and thermal hydraulic calculations undertaken in student reactor design studies. M32 is a general three-dimensional, two-energy group code that couples a diffusion theory representation of the core with a thermal hydraulic description that accounts for the heat flux and coolant enthalpy rise. A sample application examining the feasibility of a high temperature helium cooled reactor based on the General Atomics GT-MHR design for marine applications is described. M32 has been demonstrated to be a useful addition to the portfolio of computational tools for education and training in reactor design methodologies.

A Study on High-Intensity Radiation Protection of MOX-Fuel Doped with Protactinium

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The paper addresses the problem of MOX-fuel proliferation protection. The analysed measures are aimed at forming the enhanced protective radiation barrier inside of MOX-fuel during full cycle of its management. These measures include an admixture of Pa-getter into MOX-fuel composition.

Neutron irradiation converts protactinium into isotope ^{232}U, long-lived source of high-energy gamma-radiation. Being, in essence, a burnable absorber, isotope ^{231}Pa is able to decrease effectively an initial reactivity margin and, thus, to promote realization of prolonged MOX-fuel life-times.

The performed evaluations show that accumulation of uranium getter constitutes significant fraction (∼40%) of initial protactinium loading. Formation of U-Pa getter containing significant uranium fraction is an important condition for reaching high fuel burn-up.

It is demonstrated that, under irradiation of MOX-fuel with Pa-getter, a long-term radiation barrier is formed in the fuel. In case of the uranium recycle, the barrier is transferred naturally to fresh MOX-fuel. It is shown that, in the case of long-term storage of spent fuel, introduction of (Pa-U)-getter makes it possible to give a proliferation self-protection property to MOX-fuel even against short-term unauthorized actions. The preventive measures are proposed to decrease gamma-activity of nuclear materials at fabrication stage of MOX-fuel assemblies.

Optimization Studies for Seed-and-Blanket Unit (SBU) Fuel Assemblies in PWRs

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The objective of recent studies as part of a joint NERI project between Brookhaven National Laboratory and Massachusetts Institute of Technology was to develop a seed and blanket unit (SBU) fuel assembly (FA) for thorium utilization in pressured water reactors (PWR). This heterogeneous design utilizes enriched uranium seed and mixed uranium/thorium dioxide blanket fuel pins in physically separable sub-assemblies, and could be used as a one-for-one replacement of more conventional UO_2 FA's in PWRs. The seeds are used for three 18-month cycles in a 3-batch fuel core management strategy, while the blankets are used for six 18-month cycles before being removed for long-term storage; the concepts considered here assume a once-through fuel cycle consistent with current U.S. policy. The SBU should have improved proliferation, radiation safety, and comparable economic characteristics in comparison to existing PWR FA's. Optimization studies using the BOXER, COBRA-EN, and ORIGEN codes have led to an improved SBU FA design for PWRs that achieves high burnup (80 GWd/t over 3 cycles for SBU) with reduced proliferation and toxicity characteristics, while satisfying thermal-hydraulic safety limits (minimum departure from nucleate boiling ratio greater than 1.3 for 118% over-power). The SBU is an attractive fuel for existing and next-generation PWR.

8A. Fuel Cycle Physics

Session Organizers: Alain Zaetta (CEA Cadarache), Michael Todosow (BNL).
Session Chairs: Michael Todosow (BNL), John Stillman (ANL).

Reactivity and Neutron Emission Measurements of Burnt PWR Fuel Rod Samples in LWR-PROTEUS Phase II

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Fuel rods with burnup values beyond 50 GWd/t are characterised by relatively large amounts of fission products and a high abundance of major and minor actinides. Of particular interest is the change in the reactivity of the fuel as a function of burnup and the capability of modern codes to predict this change. In addition, the neutron emission from burnt fuel has important implications for the design of transport and storage facilities. Measurements have been made of the reactivity effects and the neutron emission rates of uranium oxide and mixed oxide burnt fuel rod samples having a wide range of burnup values and coming from a Pressurised Water Reactor (PWR). The reactivity measurements have been made in a PWR lattice moderated in turn with: water, a water and heavy water mixture, and water containing boron. A combined transport flask and sample changer was used to insert the 400 mm long burnt fuel rod segments into the reactor. Both control rod compensation and reactor period methods were used to determine the sample reactivities. An interesting exponential relationship has been found between the neutron emission rate and the measured reactivity.

Design and Analysis of Molten Salt Reactor Fueled by TRU from LWR

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This study assesses the feasibility of designing a finite once-through Molten Salt Reactor (MSR) fed with trans-uranium isotopes (TRU) from LWR spent fuel to be critical and to have a low peak-to-average radiation damage to graphite. The study also quantifies the transmutation effectiveness of this MSR considering the following measures: fractional transmutation of all actinides, of ^{239}Pu and of ^{237}Np and its precursors, radio-toxicity and decay-heat.

Three-dimensional design optimization with MCNP resulted in a graphite reflected critical core having a remarkably small peak-to-average graphite damage rate of 1.079. The optimal graphite-to-MS (C/MS) volume ratio is between 2 and 3. The transmutation characteristics of this reactor strongly depend on the MS feed-rate. They also depend on the C/MS ratio. There is no universally preferred neutron spectrum – to minimize radiotoxicity and decay heat, harder spectrum is preferred during the first few dozen years following discharge from the MSR as well as in the ten thousand and hundred thousand years time range. However, softer spectrum is preferred for periods in-between. Softer spectrum is also preferred for minimizing the inventory of fissile Pu and of ^{237}Np and its precursors and for maximizing the graphite lifetime.

On the Capability of SMORES to Account for Self-Shielding in Search for Maximum k_{eff}

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This study investigates the capability of SMORES – a new prototypic analysis sequence of SCALE-5, to identify the maximum k_{eff} composition of 5% enriched uranium – water systems having fixed fuel mass. The effects of resonance self-shielding are accounted for. Simplified one-dimensional slab geometry systems are considered. It is found that SMORES is capable of identifying nearly optimal lumped composition of cores fueled with low enriched fuel the k_{eff} of which can increase by increasing the resonance self shielding. The near optimal composition can be identified starting from either homogeneous or non-optimally lumped composition. The deviation of the converged lumped composition

from the absolutely optimal depends on the amplitude of density change per iteration in addition to the initial composition. Large density changes per iteration may impair convergence and result in far from optimal lumped composition.

A Theoretical Concept for a Thermal-Hydraulic 3D Parallel Channel Core Model

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A generally applicable theoretical concept for an adequate description of the 3D thermal-hydraulic single- and two-phase flow phenomena is presented. It will be applied to a 3D core simulated by parallel channels, allowing both single- (with sub-cooled water or superheated steam) and/or two-phase flow situations within these channels. The concept is realized in the experimental code HERO-X3D, concentrating in a first step on an 'artificial' BWR or PWR core which may consist of a central channel, four quadrants and (eventually) a bypass channel. The validity of the 3D representation has to be proofed by demonstrating that also at non-symmetric perturbations of the core the mass flow distribution into the different channels is chosen automatically in such a way that the pressure decrease over each channel stays equal.

The heat generation within each 'fuel element' of such a core will be simulated by 'point kinetic representation' (This can for real situations then easily be replaced by already well developed 3D core kinetic codes). The heat transport out of a fuel element will be determined from the solution of the discretized Fourier heat conduction equation.

The theoretical concept is based on three important development lines:

- The existence of a very generally applicable and in itself complete thermal-hydraulic **c**oolant **c**hannel **m**odule CCM (allowing a simulation of the steady state and transient behaviour of single- or two-phase flow within a general coolant channel),
- The separate treatment of the mass and energy from the momentum balance eqs. (avoiding thus the very time-consuming integration of 'stiff' eqs.) and
- The establishment of a procedure for the calculation of the mass flow distributions into different parallel channels based on the fact that the sum of pressure decrease terms over a closed loop must stay, despite of un-symmetric perturbations, zero.

A very detailed description of the theoretical concept is given, the development of the experimental code (HERO-X3D) is in progress. Test calculations for the case of the 'artificial' core will demonstrate the quality and validity of the entire concept.

Reactivity Effects due to Beryllium Poisoning of BR2

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This paper illustrates the impact of beryllium poisoning on reactivity variations of the Belgian MTR BR2 in SCK•CEN. A specific feature is the reactor core of HEU positioned in and reflected by a beryllium matrix - a big number of prismatic hexagonal prisms which are skew and form hyperboloidical arrangement around the central 200 mm channel H1 containing mainly beryllium plugs.

The poisoning effect of beryllium appears after irradiation of Be matrix with fast neutrons due to (n,α) – reaction on ^{9}Be and following transmutations into nuclides of ^{6}Li, ^{3}T and ^{3}He. The tritium concentration increases linearly with the energy produced [MW.days]. After reactor shutdown, the concentration of ^{3}He increases due to the decay of ^{3}T, thereby causing reactivity losses of the reactor core. In this paper we discuss reactivity effects due to poisoning by strong neutron absorbers ^{6}Li and ^{3}He following the full irradiation history of the 3rd Be-matrix and taking into account the different reactor core loads between April 1997 and December 2003.

The conducted study has highlighted the importance of the beryllium poisoning for reactivity predictions in BR2. The forecast for reactivity variations due to the accurate modeling of ^{6}Li poisoning is about ±(0.5±0.05)$ in case of strong perturbation of the flux in the central beryllium channel (location of a fuel element or strong absorber Ir). The possible biases associated with accuracy of the modelized ^{3}He poisoning after very long shutdown are within ±(0.6±0.06)$. The maximum ^{3}He antireactivity effect of about (4.0±0.4)$ has been calculated after the longest reactor shutdown T~100 days.

Session Organizers: Luiz Leal (ORNL), Mark Chadwick (LANL).
Session Chairs: Michael Dunn (ORNL), Soo-Youl Oh (KAERI).

An Assessment of ENDF/B-VI Releases Using the MCNP™ Criticality Validation Suite

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The final release of version VI of the Evaluated Nuclear Data File (ENDF/B-VI) was distributed in October 2001. That release came approximately two decades after the final release of its predecessor, ENDF/B-V. In the interim, several intermediate releases of ENDF/B-VI incorporated substantive changes to nuclear data for a large number of isotopes. This study provides an assessment of the reactivity behavior produced by the nuclear data in that final release and in three of the preceding intermediate releases, using the MCNP5 Monte Carlo code and the benchmarks in the MCNP criticality validation suite.

Relative to ENDF/B-V and to earlier interim releases of ENDF/B-VI, the final release for ENDF/B-VI produces better agreement with benchmark values of k_{eff} for some cases but worse agreement for others. In addition, poor agreement with the benchmark values for the most extreme cases remains essentially unaffected.

Collectively, the results obtained suggest that the cross sections for a number of nuclides still need improvement. Adjustments to the cross sections for ^{232}Th, ^{233}U, ^{235}U, ^{238}U, and ^{239}Pu conceivably could produce better agreement with the benchmark values of k_{eff} for several of the cases in the suite. In addition, agreement for some of the cases could be improved by retaining cross sections from earlier interim releases of ENDF/B-VI for certain nuclides over certain energy ranges.

Generation and Performance of a Multigroup Coupled Neutron-Gamma Cross-Section Library for Deterministic and Monte Carlo Borehole Logging Analysis

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As part of the IRTMBA (Improved Radiation Transport Modelling for Borehole Applications) project of the EU community's 5th framework program a special purpose multigroup cross-section library was prepared for use in deterministic and Monte Carlo oil well logging particle transport calculations. This library is expected to improve the prediction of the neutron and gamma spectra at the detector positions of the logging tool, and their use for the interpretation of the neutron logging measurements was studied. The library is based on the most recent ENDF/B-VI.8 nuclear data.

The multigroup library was applied in various deterministic and Monte Carlo codes for an oil well logging problem. It was also compared with a continuous-energy Monte Carlo calculation. No significant differences were found. The library is also successfully tested against an experimental benchmark for Oxygen.

The development of the library brought to light a format error in the ENDF/B-VI.8 data for ^{35}Cl. This error leads to partly negative multigroup photon production data, but could be corrected.

Session Organizers: Luiz Leal (ORNL), Mark Chadwick (LANL).
Session Chairs: Michael Dunn (ORNL), Soo-Youl Oh (KAERI).

On the Importance of a New Formula for the Double Differential Scattering Kernel

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The current scattering kernel treatment in NJOY is based on the light isotopes $S(\alpha, \beta)$ treatment for all materials. For heavy isotopes with pronounced resonances this is not quite applicable. In order to allow for resonance scattering in a way that provides the correct cross section the cumulative kernel over all angles is normalized according to the broadened elastic cross-section calculated by the NJOY module BROADR.

A new double differential scattering kernel developed by Rothenstein and Dagan includes explicitly the resonances of heavy isotopes in the model treatment and therefore can improve the current NJOY solution procedure. Moreover, former investigations and the current one indicate that the MCNP treatment in the Epi thermal resonant region might be inaccurate as the code applies approximations which are not always valid.

The new mathematical formalism developed by Rothenstein enabled the implementation of the modified kernel into the relevant NJOY module called THERMR. The improvements to the existing kernel in NJOY are demonstrated and preliminary results emphasize the importance of using the new formula for the scattering kernel for nuclides with strongly energy dependent scattering cross–sections. The essential steps to improve the MCNP sampling methods in the vicinity of pronounced resonances are also discussed.

Effect of Energy Self-Shielding Methods on ^{238}U for Criticality Safety Problems

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Multigroup cross sections are obtained by weighting point-wise nuclear data with a flux spectrum. For nuclides having a resonance structure, energy self-shielding calculations are performed to calculate a more detailed flux spectrum. Subsequently, self-shielded multigroup cross sections are generated. Different methods exist for energy self-shielding calculations. Among them are the Bondarenko method, the NJOY flux calculator, and the CENTRM method. The CENTRM method is a more advanced technique that utilizes both multigroup and point-wise cross sections in a one-dimensional transport calculation to solve for a point-wise flux distribution.

The method of energy self-shielding is one of the elements in a multigroup cross-section generation that may have a significant impact on the multiplication factor in criticality safety calculations. This paper compares the three aforementioned self-shielding methods applied to ^{238}U. A criticality problem having 23 cases is considered. This system includes water-moderated, low-enriched UO_2 fuel rods in square-pitched array, with a thermal flux spectrum. Multiplication factors obtained from transport calculations that use multigroup and continuous energy data are compared. It is observed that multiplication factors calculated with multigroup data containing different self-shielding methods for ^{238}U have less than 500 pcm difference with continuous energy results.

8B. Nuclear Data

Session Organizers: Luiz Leal (ORNL), Mark Chadwick (LANL).
Session Chairs: Michael Dunn (ORNL), Soo-Youl Oh (KAERI).

Development of a Methodology for Analysis of the Impact of Modifying Neutron Cross Sections

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Monte Carlo analysis of a Time-of-Flight (TOF) experiment can be utilized to examine the accuracy of nuclear cross section data. Accurate determination of this data is paramount in characterization of reactor lifetime. We have developed a methodology to examine the impact of modifying the current cross section libraries available in ENDF-6 format where deficiencies may exist, and have shown that this methodology may be an effective methodology for examining the accuracy of nuclear cross section data. The new methodology has been applied to the iron scattering cross sections, and the use of the revised cross sections suggests that reactor pressure vessel fluence may be underestimated.

The Fission Spectrum Uncertainty

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The fission spectrum uncertainty (variance covariance) matrix is needed in many nuclear data applications. Unfortunately, most modern evaluated data files, with the exception of JENDL3.3, do not include information on the fission spectrum uncertainty. In this paper, the fission spectrum relative uncertainty matrix derivation for the Watt fission spectrum is followed by a more general derivation utilizing some characteristics of any fission spectrum. In the second derivation, the Watt spectrum is used only for the calculation of sensitivity coefficients. In this work, sample fission spectrum uncertainty matrices are generated for the primary fissionable isotopes of ^{235}U, ^{238}U, and ^{239}Pu.

The TRADE Experiment: Importance of Neutron Cross-Sections for Transmutation

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Accurate neutron cross-section data is fundamental to the reliable design of any transmutation device, and, in particular, of an Accelerator-Driven System (ADS). Calculations of the behaviour of the core depend strongly on the cross-section data: parameters such as the multiplication coefficient, power densities or reactivity may vary significantly depending on the nuclear data (ND) library used. These potential discrepancies justify the need to improve the present data for several isotopes and reaction channels, for a wide range of neutron energies from thermal to high-energy.

The present work expands the analysis performed in the context of the n_TOF-ND-ADS project of the EURATOM 5th Framework Program, to other isotopes of interest such as ^{233}U, ^{243}Am, $^{244, 245}$Cm and the long-lived fission fragments (LLFFs) ^{99}Tc and ^{129}I. A direct comparison of nuclear data libraries to indicate the spread between values was performed. The paper also extends the sensitivity analysis mentioned above to moderated systems, such as TRADE (Triga Accelerator-Driven Experiment): a 1 MW Triga Reactor coupled with a 140 MeV–2 mA proton cyclotron. Study of the discrepancies in the thermal and epithermal regions is essential for the design of systems for the transmutation of LLFF (transmutation by adiabatic resonance crossing, TARC) and also important for minor actinides (MAs) for which sub-threshold fission should not be neglected.

Box-Cox Transformation for Resolving the Peelle's Pertinent Puzzle in a Curve Fitting

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Incorporating the Box-Cox transformation into a curve fitting is presented as one of methods for resolving an anomaly known as the Peelle's Pertinent Puzzle in the nuclear data community. The Box-Cox transformation is a strategy to make non- normal distribution data resemble normal distribution data. The proposed method consists of the following steps: transform the raw data to be fitted with the optimized Box-Cox transformation parameter, fit the transformed data using a conventional curve fitting tool, the least-squares method in this study, then inverse-transform the fitted results to the final estimates. Covariance matrices are correspondingly trans- formed and inverse-transformed with the aid of the law of error propagation.

In addition to a sensible answer to the Puzzle, the proposed method resulted in reasonable estimates for a test evaluation with pseudo-experimental ^{6}Li(n,t) cross sections in several to 800 keV energy region, while the GMA code resulted in systematic underestimates that characterize the Puzzle. Meanwhile, it is observed that the present method and the Chiba-Smith method yield almost the same estimates for the test evaluation on ^{6}Li(n,t). Conceptually, however, two methods are very different from each other and further discussions are needed for a consensus on the issue of how to resolve the Puzzle.

8C. Physics Code Validation

Session Organizers: Alain Santamarina (CEA Cadarache), Robert Jacqmin (CEA Cadarache), Richard Sanchez (CEA). Session Chair: Robert Jacqmin (CEA Cadarache).

Benchmark Calculation of the WIMS/RFSP for the Wolsong Nuclear Power Plants 3 & 4 Physics Measurement Data

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For the validation of the WIMS/RFSP code, benchmark calculations were performed using the physics measurement data of the Wolsong nuclear power plants 3 and 4. As was done in the previous study, the benchmark calculations were performed for the criticality, boron worth, reactivity device worth, reactivity coefficient, and flux scan. The results showed that the criticality and boron worth were underestimated by 3~6 mk and ~7%, respectively. The reactivity device worth was generally consistent with the measured data except for the strong absorbers such as the shutoff rods and the mechanical control absorber. The heat transport system temperature coefficient and flux distributions were in good agreement with the measured data.

Qualification of MCNP Coolant Void Reactivity Calculations using ZED-2 Measurements

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Several sets of fuel-substitution and reference-lattice measurements performed in the ZED-2 zero-power reactor were simulated with MCNP to test its reliability for CANDU coolant-void-reactivity calculations when used with an AECL-generated, continuous-energy data library based on ENDF/B-VI release 5. The reactivity results were analyzed to estimate the bias in coolant-void-reactivity calculations for pure lattices of fresh-natural-uranium and MOX (simulated irradiated natural uranium) CANDU fuels using an approximate methodology derived from perturbation theory. It was found that MCNP overestimates the coolant void reactivity for pure lattices of 28-element and 37-element CANDU fuel at room temperature by 0.7 to 1.7 mk, amounts which are generally consistent with corresponding bias values established for WIMS-IST lattice-cell calculations.

DYN3D Calculations for the V-1000 Test Facility and Comparisons with the Measurements

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In the framework of the EU project VALCO, measurements at the V-1000 test facility of the Russian Research Centre "Kurtschatov Institute" are used for the validation of three-dimensional neutronic calculations. Experimental results of steady states and kinetic experiments are available for comparisons with calculations. Respective DYN3D calculations have been performed by using the cross section libraries generated with the codes HELIOS, CASMO and WIMS.

Results obtained by the DYN3D code are compared with steady-state measurements for two different configurations of the V-1000 facility. Pin-power distributions measured within one fuel assembly are compared to the results of the pin power reconstruction implemented in DYN3D.

Two kinetic experiments performed in the V-1000 facility were simulated by the DYN3D code. Comparisons of the calculated results with the measurements at the in-core detector positions are given. The calculated fast flux of the nodes, situated near to the ionisation chambers of the two reactimeters, is compared with the detector signal. The results of the relative detector rates coincide with the measurements. The dynamical reactivities obtained from the core-averaged fluxes by inverse point kinetics are compared with reactivity curves of the two reactimeters and the calculated rod worth.

Session Organizers: Alain Santamarina (CEA Cadarache), Robert Jacqmin (CEA Cadarache), Richard Sanchez (CEA). Session Chair: Robert Jacqmin (CEA Cadarache).

Validation of SCALE4.4a/CSAS25 for Nuclear Criticality Safety Analyses

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This paper presents calculation results used for establishment of bias, bias trends and uncertainty for validation of the CSAS25 control module from the Standardized Computer Analyses for Licensing Evaluation (SCALE) Version 4.4a, using the 238-group ENDF/B-V cross-section library. Over 1200 benchmark cases were used from Volumes I through VI of the "International Handbook of Evaluated Criticality Safety Benchmark Experiments," published by the Nuclear Energy Agency Organization for Economic Cooperation and Development (NEA/OECD). Associated upper subcritical limits are also discussed.

The Nuclear Heating Calculation Scheme for Material Testing in the Future Jules Horowitz Reactor

Nicolas Huot[1], Alain Aggery[2], David Blanchet[1], Arnaud Courcelle[1], Sébastien Czernecki[3], Jacques Di-Salvo[1], Christoph Döderlein[1], Hugues Serviere[1], Guy Willermoz[1]

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The calculation of nuclear heating is an important issue for the design of the future material testing reactor called "Jules Horowitz Reactor" (JHR) and its test device.

In order to ensure a heat gradient below 5°C across the material testing sample, the deposited energy, mainly due to photon interaction, has to be calculated in various irradiation devices with an accuracy better than 15%.

To achieve this aim, an innovative heating calculation scheme has been carried out at CEA. The classical scheme chaining deterministic and stochastic code is modified. First a heterogeneous gamma source calculation is performed at assembly level using the deterministic code APOLLO2, followed by a Monte Carlo gamma transport calculation in the whole core using the TRIPOLI4 code. The calculated gamma sources at the assembly level are applied in the whole core simulation using a weighting based on power distribution obtained from the neutronic core calculation.

These gamma sources at the assembly level can be used in various core configurations at the Monte Carlo gamma transport step without repeating the complete photonic calculation.

This new scheme has been motivated by the special geometry of the JHR assembly. Moreover, the neutronic code scheme is fully validated whereas previous schemes determine gamma source on homogeneized core resolution without intermediate neutronic validation.

This new heating scheme has been validated by comparison with a 3D Monte Carlo calculation in coupled neutron-gamma mode, which gives a result without any assumption (except concerning the nuclear data used), but with a run-time consuming 4 to 5 times greater to reach the same precision.

The discrepancies between the developed scheme and the reference calculation for a simple experimental device (single CHOUCA) are below 3%, which is close to the Monte Carlo statistical uncertainty.

A series of mock-up experiments in the EOLE research reactor is going to furnish a set of gamma dose measurements, allowing to consolidate the validation.

8C. Physics Code Validation

Session Organizers: Alain Santamarina (CEA Cadarache), Robert Jacqmin (CEA Cadarache), Richard Sanchez (CEA). Session Chair: Robert Jacqmin (CEA Cadarache).

Analysis of the ZPR-9 Gas-Cooled Fast Reactor Experiments Using JEF-2.2 Data and the ERANOS Code System

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The recent renewal of interest for gas-cooled fast reactors has led CEA to revisit past experiments for code validation purposes. Several configurations of the first phase of the gas-cooled fast reactor experiments performed in 1975-76 in the Zero Power Reactor 9 critical facility at Argonne National Laboratory have been reanalyzed using the ERANOS-2.0 neutronics code system and JEF2-based unadjusted and adjusted nuclear data libraries. Some numerical validations have been performed with the Monte-Carlo code TRIPOLI-4. The reactivity scale (delayed neutron fraction, inhour value) is predicted within the experimental uncertainty. The criticality of the core is correctly predicted (discrepancy <0.5% when all model biases have been taken into account) even if the calculated magnitude of the heterogeneity and streaming effects, although small, differs significantly between the deterministic (-0.46%) and stochastic (-0.22±0.03%) calculations. Fission spectral indices relative to Pu239 are well predicted (discrepancy <2%), with the exception of Pu240 (overestimation of 9% with unadjusted data, reduced to 5% with adjusted data), but capture indices (Th232, U238) are underestimated by ≈8 %. For U238, this is inconsistent with the reactivity prediction. It is believed that a better description of the detailed structure and position of these foils could help in reducing the discrepancy.

Spatially and Temperature Dependent Dancoff Method for LWR Lattice Physics Code

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Spatially and temperature dependent self-shielding method SDDM was developed, and validated by comparing its reactivity changes due to fuel temperature with those obtained from the continuous Monte Carlo code calculation.

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Cecil Parks (ORNL).

French Calculation Schemes for Light Water Reactor Analysis

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This paper describes the recent code APOLLO2.5/CEA93.V6 currently implemented in the LWR calculation tools of CEA, Framatome-ANP and Electricité de France. The corresponding packages SAPHYR, SCIENCE.V2 and CASSIOPEE are presented. The recommended *"CEA-97"* reference scheme is detailed and some results of the Validation/Qualification work are enhanced. It is shown that the major design parameters are predicted within the required target accuracy in a wide experimental validation range: up to 4.8% w/o ^{235}U in UO_2 fuel and 13%Pu in MOX assemblies, for burnup ranging from 10 to 60 GWd/t.

Irradiation Experiment Analysis for Cross Section Validation

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An analysis of the irradiation experiment TRAPU performed at the French fast reactor PHENIX is presented. The observed C/E's on the final densities of the measured isotopes for the different basic data files (JEF2.2, ENDF/B-V, and ENDF/B-VI) indicate some large discrepancies that a subsequent sensitivity analysis attributes to specific cross sections of actinides. Among them: ^{238}U σ(n,2n), and capture cross sections of higher plutonium isotopes, ^{241}Am, ^{243}Am, and ^{242}Cm. capture cross sections. Very useful information is gained by performing a simulated adjustment that uses the observed C/E's and the calculated sensitivity coefficients. In particular this adjustment shows that a good consistency can be reached on final adjusted cross sections, and associated new C/E's, starting from different data files. The same adjustment shows that we cannot reach definite conclusions on many fission cross sections (^{238}Pu, ^{241}Pu, ^{242}Am, and ^{243}Cm), because the experiments were not sensitive enough to these parameters.

Methodologies for Treatment of Spectral Effects at Core-Reflector Interfaces in Fast Neutron Systems

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The existence of space, angle and energy neutron flux distribution transients is a well known phenomenon which makes the calculation of fast reactors difficult when the core is surrounded by reflectors, particularly steel reflectors. The current interest for fast neutron reactors as TRU burners of relatively low power and small core size (high leakage) motivates the search for an accurate deterministic treatment of neutron reflection. Indeed, without due care taken, the treatment of reflector effects can introduce significant uncertainties in K_{eff} and power distribution calculated values.

We report here further investigations that we have performed with the deterministic code system ERANOS2.0 and with the continuous energy Monte Carlo code TRIPOLI-4, in order to understand these effects and converge to a satisfactory calculation scheme for ERANOS: our final objective is to define a recommended deterministic calculation procedure for an accurate treatment of fast reactors neutron balance in the presence of reflectors.

Our work shows that satisfactory results can be achieved by optimization of the macrocell option already implemented in the ECCO cell code of ERANOS, the agreement with the Monte Carlo calculation becoming satisfactory even when using a reduced number of energy groups.

8D. Reactor Analysis Methods

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Cecil Parks (ORNL).

Effect of Pellet Radial Power and Temperature Distributions on Fuel Assembly Neutronics

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Recently, the resonance self-shielding module of the PARAGON code has been enhanced with the capability of subdividing the fuel pellet into multi-regions. A gradient of temperature can be associated to these micro-regions to better account for the spatial temperature distribution. The purpose of this paper is to validate this module when a profile of temperature is used in the fuel pellet. We will compare the results to the current core design methodology where a flat temperature is used throughout the pellet. The profile of temperature is generated by the Westinghouse fuel rod design code PAD. Since the radial temperatures depend on the radial powers and vice-versa, iteration between PARAGON and PAD is necessary.

In the first step we will benchmark the code against the continuous energy Monte Carlo MCNP calculation for different types of fuel, like UO2, MOX, annular fuel etc.

The paper will then focus on the comparison between flat temperature model and temperature gradient model both for the single pin cell and the assembly cases. The study will include different fuel types: different cell pitches, different enrichments, different burnable absorbers, etc. The neutronic parameters of interest in this study will be the reactivity and pin power distribution from BOL to EOL.

A Comparison of Binary Stochastic Media Transport Models in “Solid-Void” Mixtures

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We provide numerical comparisons of the Levermore-Pomraning and atomic mix transport models with benchmark calculations for a class of planar geometry binary stochastic media systems in which one of the two materials is a void. We show that the Levermore-Pomraning approximation is always the superior approximation for low to moderate scattering ratios in the solid material, and is competitive with atomic mix in diffusive systems of this class. We also show trends in the error of the two approximations as well as trends in the standard deviation of the benchmark ensemble-average scalar flux.

Development of an Object Oriented Nodal Code using the Refined AFEN Derived from the Method of Component Decomposition

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In this study, we invented a method of component decomposition to derive the systematic internodal coupled equations of the refined AFEN method and developed an object oriented nodal code to solve the derived coupled equations. The method of component decomposition decomposes the intranodal flux expansion of a nodal method into even and odd components in three dimensions to reduce the large coupled linear system equation into several small single equations. This method requires no additional technique to accelerate the iteration process to solve the internodal coupled equations, since the derived equations can automatically act as the coarse mesh rebalance equations. By utilizing the object oriented programming concepts such as abstraction, encapsulation, inheritance and polymorphism, dynamic memory allocation, and operator overloading, we developed an object oriented nodal code that can facilitate the input/output and the dynamic control of the memories, and can make the maintenance easy.

8D. Reactor Analysis Methods

Session Organizers: Giuseppe Palmiotti (ANL), Giovanni Bruna (Framatome ANP), Jasmina Vujic (UC Berkley).
Session Chair: Cecil Parks (ORNL).

Applications of Modal-Local Analysis for Source-Driven Subcritical Systems

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The modal-local method commonly used for critical systems has been adapted for space-time analysis of source-driven subcritical systems. The flux is represented as a sum of a local component describing the direct perturbations due to variations of system parameters and a global function determining the overall flux changes. The global function is expanded in terms of generalized λ-modes accounting for the presence of the source in the system. The method was applied to dynamic analysis of the reactor response to a step source change. Comparison with time-dependent ERANOS simulations indicates the applicability of the proposed method even with a few expansion terms for the global function.

Development of a Polynomial Nodal Model for the Multigroup Diffusion Equation in 2-D

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A nodal model which uses Legendre polynomials as the expansion functions is developed for the two dimensional multi-group diffusion equation solutions. In this model, the neutron flux is expanded as the Legendre polynomials in both of the spatial variables. The coefficients of the unknown polynomials are calculated from a set of linear equations. These equations are obtained as a result of applying the least-squares method and the use of the interface conditions. The average one dimensional partial currents are extended to the two dimensional case and used as the interface conditions. This model is based on an iterative process in which the unknown polynomial coefficients are evaluated from a set of linear equations which obtained from the least-squares minimization of the polynomial expansion residuals and the interface conditions. Once the polynomial coefficients are found in a given iteration, new interface conditions and the eigen-value are calculated. The iteration process is then continued until two given convergences criterions are satisfied for eigen-value and eigen-function. The IAEA-2D benchmark problem is used to check the accuracy of presented nodal model. The results are compared with finite-difference and Green's Function nodal method solutions of the two dimensional diffusion equation and shows that the model compared favorably.

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	Adams, James M.	*Monte Carlo Modeling of a Time-of-Flight (ToF) Experiment for Determination of Fe Scattering Cross Sections*	3A
	Adams, Marvin	*Capturing the Effects of Unlike Neighbors in Single-Assembly Calculations*	4A
	Adomavicius, Arvydas	*The Studies of RBMK-1500 Reactor Core Behavior during Abnormal Operation Transients*	4A
	Aggery, Alain	*HORUS3D Code Package Development and Validation for the JHR Modeling*	6D
	Aggery, Alain	*Method of Characteristics Applied to a MTR Whole Core Modeling*	1D
	Aggery, Alain	*The Nuclear Heating Calculation Scheme for Material Testing in the Future Jules Horowitz Reactor*	8C
	Agostini, Pietro	*The TRADE Experiment: Status of the Project and Physics of the Spallation Target*	2B
	Agung, Alexander	*Passive Decay Heat Removal in a Fluidized Bed Nuclear Reactor*	7A
	Aisen, Elyaz M.	*An Influence of Core Physics Peculiarities upon the Thermal Hydraulics Performance in Cascade Subcritical Molten Salt Reactor*	4A
	Akiba, Miyuki	*Experimental Study on Reduced Moderation BWR with Advanced Recycle System (BARS)*	6A
	Alcock, J. P.	*A Three Dimensional Two Energy Group Coupled Reactor Physics and Thermal Hydraulics Code (M32) - A Tool for Student Design Studies*	7D
	Aldama, Daniel L	*Generation and Performance of a Multigroup Coupled Neutron-Gamma Cross-Section Library for Deterministic and Monte Carlo Borehole Logging Analysis*	8B
	Alekseev, Pavel N.	*Influence of Nuclear Fuel Cycle Duration and Reprocessing Losses Level on the Nuclear Power System Structure*	3C
	Alekseev, Pavel N.	*Studies of Physical Features of Cascade Subcritical Molten Salt Reactor with External Neutron Source*	7A
	Aliberti, Gerardo	*Methodologies for Treatment of Spectral Effects at Core-Reflector Interfaces in Fast Neutron Systems*	8D
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	Aliberti, Gerardo	*Reactivity Assessment and Spatial Time-Effects from the MUSE Kinetics Experiments*	3A
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	Alonso, Gustavo	*Solution of the 1D Kinetic Diffusion Equations Using a Reduced Nodal Cubic Scheme*	4A
	Alpan, F. Arzu	*Effect of Energy Self-Shielding on Reactor Benchmark Problems*	8B
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	Al-Qasir, Iyad I.	*Thermal Neutron Scattering Cross Sections of Thorium Hydride*	2C
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	Andristos, Dimitrios	*AGENT Code: Open-Architecture Analysis and Configuration of Research Reactors – Graphic Tools*	4A
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	Anistratov, Dmitriy Y	*Low-Order Quasidiffusion Equations in 2D Geometry*	2D
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	Antony, Muriel	*MINERVE Reactor Characterization in Support of the OSMOSE Program: Spectral Indices*	1A
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	Apse, Vladimir Alexandrovich	*High Fuel Burn-Up and Nonproliferation in PWR-Type Reactor on the Basis of Modified Th-fuel*	5A
	Arakawa, Yasushi	*Development of Hybrid Core Calculation System Using 2-D Full-Core Heterogeneous Transport Calculation and 3-D Advanced Nodal Calculation*	5D
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	Avdic, Senada D.	*Support Vector Machine in Classification of Positron Lifetime Spectra*	2C
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	Bae, Sung-Man	*Application of the Dynamic Control Rod Reactivity Measurement Method to Korea Standard Nuclear Power Plants*	3A
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	Baker, Steven	*TransLAT Lattice Physics Code Benchmark to the B&W Gadolinia Criticals*	3D
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	Bathke, Charles G.	*Dynamic Analysis of the AFCI Scenarios*	3C
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	Bejmer, Klaes-Håkan	*Enriched Gadolinium as Burnable Absorber for PWR*	5A
	Belousov, Anton	*The Studies of RBMK-1500 Reactor Core Behavior during Abnormal Operation Transients*	4A
	Berger, H.-D.	*Reactivity and Neutron Emission Measurements of Burnt PWR Fuel Rod Samples in LWR-PROTEUS Phase II*	8A
	Bernard, David	*Experimental Validation of APOLLO2 Code for High Burnup Mox Fuels. JEF2.2 Results and JEFF3.0 Improvements*	6C
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	Billebaud, Annick	*Reactivity Measurements and Neutron Spectrometry in the MUSE-4 Experiment*	5C
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	Brenizer, Jack	*Thermal Neutron Time-of-Flight Spectroscopy at Penn State using a Single-Disk Chopper*	7D
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